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Non-proliferation characteristics of nuclear energy and radiological assessment of a near-surface deposit

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Abstract

Nuclear proliferation is a worldwide issue: even though the number of nuclear weapons has decreased in time they still represent a menace to the world peace. This problem can be addressed by implementing an efficient, advanced and constant system of monitoring of nuclear material, the safeguards, which ensures the peaceful use of nuclear technology. This process must be guaranteed at all stages of the nuclear fuel cycle, from the front-end to the back-end. In particular, nuclear decommissioning is one of the most complicated phases due to the many fields and activities involved. A crucial step is the realization of a deposit for radioactive waste which must be radiologically safe for the people and the environment.

This thesis aims to provide an overview of the main non-proliferation and decommissioning issues and then to analyze the radiological impact of an hypothetical near-surface deposit in Italy. Firstly, the characteristics of the nuclear safeguards and decommissioning are examined, focusing on the situation in Italy. It has been observed that, thanks to the application of innovative technologies, the facilities should be safely and securely decommissioned around 2030s and the construction of the Deposito Nazionale for the safe and secure disposal of Italian radioactive waste is currently underway, with new advances right this year. After that, a radiological safety assessment for a plausible scenario for Itay using the code RESRAD (RESidual RADioactivity) will be performed in order to evaluate the impact of some relevant radionuclides. The results show that the total dose never exceeds the 1 $\frac{mSv}{y}$ limit for the public.

Abstract

La proliferazione nucleare è un problema mondiale: anche se il numero di armi nucleari è diminuito nel tempo, rappresentano ancora una minaccia per la pace mondiale. Questo problema può essere affrontato implementando un efficiente, avanzato e costante sistema di monitoraggio del materiale nucleare, le safeguards, che assicuri l'uso pacifico della tecnologia nucleare. Questo processo deve essere garantito in tutte le fasi del ciclo del combustibile nucleare, dal front-end al back-end. In particolare, il decommissioning nucleare è una delle fasi più complicate per i numerosi campi e attività coinvolti. Un passaggio cruciale è la realizzazione di un deposito per i rifiuti radioattivi che deve essere radiologicamente sicuro per la popolazione e per l'ambiente.
Con questa tesi si vuole fornire una panoramica dei principali problemi legati alla non proliferazione e al decommissioning e quindi di analizzare l'impatto radiologico di un ipotetico deposito semisuperficiale in Italia. In primo luogo, vengono esaminate le caratteristiche delle safeguards e del decommissioning nucleari, soffermandosi sulla situazione in Italia. È stato osservato che, grazie all'applicazione di tecnologie innovative, gli impianti dovrebbero essere dismessi in sicurezza intorno al 2030 ed è attualmente in corso la costruzione del Deposito Nazionale per lo smaltimento in sicurezza dei rifiuti radioattivi italiani, con nuovi progressi proprio in questo anno. Successivamente, verrà eseguita una valutazione della sicurezza radiologica per uno scenario plausibile per l'Italia utilizzando il codice RESRAD (RESidual RADioactivity) al fine di valutare l'impatto di alcuni radionuclidi rilevanti. I risultati mostrano che la dose totale non supera mai il limite di 1 $\frac{mSv}{a}$ per il pubblico.

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Acronyms

3DLR	3-Dimensional Laser Rangefinder
AP	Additional Protocol
BWR	Boiling Water Reactor
CA	Complementary Access
CCI	Compton Compact Imager
CEMEX	CEMentazione EurEX
CIPE	Comitato Interministeriale per la Programmazione Economica
CNAI	Carta Nazionale delle Aree Idonee
CNAPI	Carta Nazionale delle Aree Potenzialmente Idonee
CSA	Comprehensive Safeguards Agreement
C/S	Containment and Surveillance
DA	Destructive Analysis
DCVD	Digital Cherenkov Viewing Device
DIE	Design Information Examination
DIQ	Design Information Questionnaire
DIS	Digital Image Surveillance
DIV	Design Information Verification
DOE	Department Of Energy
EACA	European Association of Competent Authorities for the Safe Transport of Radioactive Material
ENEA	National Agency for New Technologies, Energy and the Environment
EPA	Environmental Protection Agency
EU	European Union
Euratom	European Atomic Energy Community
EUREX	Enriched URanium EXtraction
EW	Exempted Waste

FRMF	Free Release Measurement Facility
GPR	Ground Penetrating Radar
HDPE	High Density PolyEthylene
HERCA	Heads of the European Radiological Protection Competent Authorities
HEU	Highly Enriched Uranium
HLW	High Level Waste
HVAC	Heating, Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
ICPF	Impianto Cementazione Prodotto Finito
ICRP	International Commission on Radiological Protection
IDMS	Isotopic Diluition Mass Spectrometry
ILW	Intermediate Level Waste
INES	International Nuclear and radiological Event Scale
IRAT	IRradiated Attribute Tester
ISIN	Ispettorato nazionale per la sicurezza nucleare e la radioprotezione
ISOCS	In Situ Object Counting System
ISPRA	Istituto Superiore per la Protezione e la Ricerca Ambientale
ITREC	Impianto di Trattamento e Rifabbricazione Elementi di Combustibile
JRC	Joint Research Center
KORAD	KORea RADioactive waste agency
LECO	Latina Estrazione COndizionamento
LEU	Low Enriched Uranium
LLW	Low Level Waste
LWR	Light Water Reactor
MOX	Mixed OXide
NDA	Non Destructive Analysis
NE	Neutron Emission
NGAT	Neutron and Gamma Attribute Tester
NIWAS	Nucleco Integrated Waste Assay System
NMA	Nuclear Material Accountancy
NML	Nuclear Material Laboratory
NNWS	Non-Nuclear Weapon State
NPT	Nuclear Non-Proliferation Treaty
NRC	Nuclear Regulatory Commission

NWAL	NetWork of Analytical Laboratories
NWAS	Nucleco Waste Assay System
NWS	Nuclear Weapon State
OECD	Nuclear Energy Agency at the Organization for Economic Cooperation and Development
PANWAS	Passive Active Neutron Waste Assay System
PGET	Passive Gamma Emission Tomography
PIV	Physical Inventory Verification
PWR	Pressurized Water Reactor
RATA	Radioactive waste management agency
RESRAD	RESidual RADioactivity
SAL	Safeguards Analytical Laboratory
SFAT	Spent Fuel Attribute Tester
SiCoMoR	Sistema di Condizonamento Modulare Rifiuti
SIRIS	SIstemazione RIfiuti Solidi
SMOPY	Safeguards MOX PYthon
SOGIN	Società per la Gestione degli Impianti Nucleari
SQA	Small Quantity Protocol
SSNC	Small Sample Neutron Counter
SRWGA	Sea Radioactive Waste Gamma Analyser
TGS	Tomographic Gamma Scanner
TIMS	Thermal Ionization Mass Spectrometry
UOX	Uranium OXide
VLLW	Very Low Level Waste
VOA	Voluntary Offer Agreement
VSLW	Very Short-Lived Waste
WAC	Waste Acceptance Criteria
WENRA	Western European Nuclear Regulators Association
WIPP	Waste Isolation Pilot Plant
WMF	Waste Management Facility
WOT	Wet Oxidation Technology

Chapter 1

Introduction

One of the most critical tasks facing the world is to establish and keep controls of nuclear fuels and radioactive materials all over the world to avoid any destructive use. The world's nuclear-armed states possess a combined total of about 13,080 nuclear warheads [1] which is a much lower number with respect to the past but it still quite impressive. Thus, nuclear material must be continuously monitored to guarantee it is used only for peaceful purposes. This is the reason why the International Atomic Energy Agency (IAEA) has developed a safeguards system which an huge set of technical measures by which the IAEA verifies the correctness and the completeness of the declarations made by States about their nuclear material and activities. Together with modern and sophisticated tools, the IAEA stipulates agreements with the States in order to apply an efficient safeguards strategy.
This issue must be covered all over the nuclear fuel cycle, including the back-end. About 115 commercial power reactors, 48 experimental or prototype power reactors and 250 research reactors and several fuel cycle facilities, have been retired from operation. In particular, of the more than 160 power reactors, including experimental and prototype units, at least 17 have been fully dismantled. [2] Nuclear decommissioning is a challenging task since many experts of many fields are involved in this activity, from the technical side to the economical one, passing through the social aspects, but also security: all the spent fuel and the nuclear material still present must be safeguarded in order to reduce the risks of purposeful destructive uses of these materials.
Even though Italy has not a nuclear program, it is not exempt from these issues since its nuclear facilities must be decommissioned safely and respecting the non-proliferation regime. Besides, Italy still does not have a final deposit for its radioactive waste, but the development of a facility called Deposito Nazionale for their disposal and of a research center called Parco Tecnologico is currently ongoing. This kind of facility needs to be radiologically safe for the population and the environment, thus the dose limits must be respected and efficient tools must be implemented.
The objective of this thesis is to provide an overview of the technical aspects of the safeguards and of the nuclear decommissioning, with a focus on the Italian situation, and to perform a radiological safety assessment of a near-surface deposit.
Firstly, a description of the main safeguards tools and agreements, and of the nuclear decommissioning activity, as well as of the radioactive waste management, will be given. The next chapter will show how the safeguards are applied during the decommissioning and how

the spent fuel measurements are performed. The successive chapter will get into the details of the Italian condition, by describing the procedure and the status of the decommissioning of the nuclear facilities, the situation and the future role of the Deposito Nazionale and the Parco Tecnologico under construction and the national institutions that deal with these topics. The following chapter will describe RESidual RADioactivity (RESRAD), a code which is widely used in the nuclear decommissioning area, and it will be applied to an already performed radiological assessment to show its reliability. In the subsequent chapter a radiological safety assessment of a theoretical near-surface deposit suitable for Italy will be performed and the results will be shown. Finally, in the last chapter the conclusions drawn will be presented.

Chapter 2

Nuclear safeguards and decommissioning

2.1 Proliferation

Nuclear proliferation refers to the spread of nuclear weapons, fissile materials, and weapons-applicable nuclear technology, and it is able to destabilize international or regional relations or infringe upon the national sovereignty of states due to the mutual ensured destruction situation equilibrium: if you think to attack a Nuclear Weapon State (NWS) or its allies it can respond before being destroyed and it can ensure your destruction.
Time and resources required to make fission bombs depend on the kind of explosive wanted. Correct information, skilled people, nuclear and non-nuclear materials and components are required.
The information required for the design and construction of fission explosives is available in the open literature, so the performance of the bombs designed by a group will depend on their access to information about certain concepts that are now classified, or on their levels of understanding, or both.
Three kinds of people can be useful to develop and build nuclear explosives: people with direct experience in designing, building, or testing nuclear explosives, people with highly developed technical skills and basic knowledge of the specific technical fields required for such a project, and people with necessary basic skills, but without specific knowledge or experience in the specific fields required for the project. [3]
There are thousands of people in the first category, concentrated in those nations that have built and tested nuclear weapons. There are at least tens of thousands of people in the second category, distributed among the industrialised nations, especially in those nations with extensive programs for research and development of nuclear technology for civilian purposes. There are millions of people in the last category-scientists, engineers, and technicians with education and experience in the physical sciences and engineering.
Three types of fissionable materials that are used or produced in components of civil nuclear technology can be used to develop fission explosives: Highly Enriched Uranium (HEU), plutonium, and ${}^{233}_{92}U$.

The critical masses of spheres of metallic uranium (density 19 $\frac{g}{cm^{-3}}$) surrounded by a 15-cm thick reflector of natural uranium, for different levels of $^{235}_{92}$U enrichment, are shown in Table 2.1. For comparison, the critical masses of 100% $^{235}_{92}$U spheres inside 15-cm thick reflectors of aluminium, water, nickel, and beryllium are 28, 23, 20, and 11 kg, respectively.

Enrichment of U-235 in percentage	**Critical mass of U-235 [kg]**	**Critical mass of U [kg]**
100	15	15
80	17	21
60	22	37
40	30	75
20	50	250
10	130	1300

Table 2.1. Critical mass of uranium vs U-235 enrichment. [3]

HEU is often taken to refer to uranium enriched above 20% in $^{235}_{92}$U, and has been used by the U.S. Atomic Energy Commission as a cut-off in enrichment, below which it was considered that further enrichment would be required to make a practical bomb.
Plutonium with all isotopic compositions could be in principle used as the core material for fission explosives. The spherical critical mass of α-phase (density 19 $\frac{g}{cm^{-3}}$) $^{239}_{94}$Pu in a thick natural uranium reflector is 4.4 kg. The critical mass of plutonium metal as a function of the contained isotopes that are not fissionable by thermal neutrons ($^{240}_{94}$Pu and $^{242}_{94}$Pu) can be estimated from simple diffusion theory corrected to give results that agree with measured critical masses. The results of this estimate are shown in Table 2.2.

Volume fraction of Pu-240 + Pu-242	**Critical mass of Pu-239 in a thick U reflector [kg]**	**Total Pu critical mass [kg]**
0	4.4	4.4
10	4.5	5.0
20	4.5	5.6
30	4.6	6.7
40	4.7	7.8
50	4.8	9.6

Table 2.2. Critical mass of plutonium vs isotopic composition. [3]

The volume fractions of non-fissile plutonium used in the nuclear industry vary over huge ranges, from less than 10 % for plutonium produced in fast breeder reactor blankets to $20 - 35\%$ for plutonium produced in light water reactors and $40 - 50\%$ for plutonium in fast breeder reactor cores.
The presence of $^{240}_{94}$Pu and $^{242}_{94}$Pu can affect the performance of fission explosives because

these two isotopes have also some probability of decaying by spontaneous fission. The spontaneous fission rates are of the order of $10^6 \frac{neutrons}{kg\ s}$. Neutron generation times in typical fission explosives are of the order of $10^{-8}\ s$, and something like 40-50 generations of fissions are required to build up a fast chain reaction to an explosive level. It is therefore clear that, at least under some conditions, such a neutron source would make an assembly explode before it has reached maximum criticality.
As a material for use in fission explosives, $^{233}_{92}$U is more similar to plutonium than $^{235}_{92}$U, with one important exception: the neutron release rates from $^{233}_{92}$U in the form it has when it is extracted from the reprocessing process are several orders of magnitude lower than the ones from typical forms of plutonium used in reactors. The critical mass of $^{233}_{92}$U metal or compounds is about 30% greater than that of the same chemical forms of α-phase $^{239}_{94}$Pu, or about 5.8 kg of metal in a thick uranium reflector.
Given the required fissionable materials, knowledge and skilled people, the additional (non-nuclear) materials, equipment, and facilities required to make fission explosives can vary over a wide range of degrees of accessibility and complexity, depending on the desired explosive characteristics, the degree of concern for the safety of the people involved, the time available to complete the process, the need for secrecy of the operations, and many other factors.
New national programs to develop and build fission weapons with characteristics specifically adapted to national needs could now result in a huge variety of very sophisticated types of nuclear explosives. Much of the required work could be done before the nuclear core materials were made available. If it were very important for a nation to do so, it could acquire militarily useful fission weapons for which design, construction, and non-nuclear tests had been carried out in advance, and arm them within a few days of the time when it gained access to the required amounts of concentrated fissionable materials. [3]

2.2 IAEA Safeguards

In 1957 the IAEA was founded with the purpose of encouraging and assisting research, development and practical application of nuclear energy for peaceful uses throughout the world. It is currently formed by 173 Member States and it establishes and administers safeguards designed to ensure that nuclear energy is not used for military purposes.
The IAEA's policy-making bodies comprise the General Conference of all Member States and the 35-member Board of Governors. [4] The General Conference consists of representatives of the IAEA Member States and meets in a regular annual session to consider and approve the IAEA's budget and to decide on other issues raised by the Board of Governors, the Director General and Member States. [5] The Board of Governors examines and makes recommendations to the General Conference on the IAEA's financial statements, programme and budget. It considers applications for membership, approves safeguards agreements and the publication of the IAEA's safety standards, and appoints the Director General of the IAEA, with the approval of the General Conference. [6]
The safeguards are a system of inspection and verification of the peaceful uses of nuclear materials as part of the Nuclear Non-Proliferation Treaty (NPT), supervised by the IAEA. They is a set of technical measures applied by the IAEA on nuclear material and activities, through which the Agency independently verifies that nuclear facilities are not misused and

nuclear material is not diverted from peaceful uses. States accept these measures through the conclusion of safeguards agreements.
IAEA safeguards are an essential component of the international security system. According to the Article 3 of the NPT, each Non-Nuclear Weapon State (NNWS) is required to conclude a safeguards agreement with the IAEA. The Agency negotiates safeguards agreement with each State and verifies the State's compliance. [7] If a non-compliance occurs it may be reported to the United Nations Security Council.
The implementation of safeguards follows an annual cycle and comprises four main processes:

1. Collection and evaluation of safeguards-relevant information: the IAEA collects, processes and reviews all available relevant information about a State to evaluate its consistency with its declarations about its nuclear programme.

2. Development of a safeguards approach for a State: a safeguards approach for a State includes those safeguards measures to achieve the technical objectives for verifying the State's declarations.

3. Planning, conducting and evaluating safeguards activities: the IAEA develops a plan specifying the safeguards activities to be conducted both in the field and at the Agency's headquarters. Once an activity has taken place, the IAEA evaluates the extent to which it has reached the technical objectives and identifies any inconsistencies that might need to be followed up.

4. Drawing of Safeguards conclusions: the safeguards conclusions drawn by the IAEA are based on its independent verification and findings. They are the final product of the annual safeguards implementation cycle and provide credible assurances to the international community that States are abiding by their safeguards obligations. [8]

International safeguards are made of many components that will be explained later.

2.2.1 Nuclear Material Accountancy

Nuclear Material Accountancy (NMA) is the part of the safeguards that is related to systems and procedures to prepare and maintain accounting records, perform measurements, and to analyse the information in order to confirm the presence of nuclear materials and to detect potential theft, loss, or diversion of nuclear materials. The main strength of material accounting is its capability of detecting anomalies and providing assurance for the operating system that nuclear materials are present in the absence of significant anomalies. Furthermore, performing physical inventories and drawing material balances provide an exacting cross-check on the overall effectiveness of a facility's material controls and on the absence of unidentified loss mechanisms that could include theft or diversion. [9]
IAEA inspectors make independent measurements to verify quantitatively the amount of nuclear material presented in the State's accounts. They count items (fuel assemblies, bundles, rods, containers of powdered compounds of uranium or plutonium, etc.), measure attributes of them during their inspections through Non-Destructive Analysis (NDA) techniques, and compare their discoveries with the declared figures and the operator's records with the purpose of detecting missing items. The next level of verification aims to detect

whether a fraction of a declared amount is missing (partial defect test) and may involve the weighing of items and measurements using NDA techniques capable of measuring an amount of nuclear material with a very high accuracy. For detecting bias defects, which would arise if small amounts of material were diverted over a protracted length of time, it is necessary to sample some of the items and to apply physical and chemical analysis techniques of the highest possible accuracy, typically more than 99%. In order to apply these Destructive Analysis (DA) techniques, the IAEA requires access to laboratories that employ such accurate techniques on a routine basis. [10]
More details and some examples of NDA and DA techniques will be given in the second chapter.

2.2.2 Containment and Surveillance

C/S techniques are applied to supplement NMA by providing means by which access to nuclear material can be controlled and any undeclared movement of nuclear material can be detected. C/S techniques are extensively used because they are flexible and cost effective: they reduce inspection costs and the level of intrusiveness of the IAEA into normal operational activity of nuclear facilities under safeguards. Moreover, C/S measures are applied in a systematic manner to monitor all diversion paths considered credible at the boundary of a facility, to ensure that transfers of nuclear material take place only at declared key measurement points. This application becomes increasingly important in large facilities where the IAEA's quantitative safeguards goals are difficult to realize exclusively through conventional NMA measures. [11] [10]
A sealing system consists of three components: a containment enclosing the nuclear material to be safeguarded, a means of applying the seal (like a metal wire) and the seal itself. All of them must be examined in order to verify that a sealing system has fulfilled its function. Seals are tamper-indicating devices used to secure materials, documents, data signals or any other important items. If designed properly, a sealing system is able to provide evidence of any unauthorized attempt to gain access to secured material. Besides, seals also provide a means of uniquely identifying secured containers. Depending on the kind of application, several seals are in use by the IAEA. However these seals do not provide any kind of physical protection. Since so much reliance is placed on sealing systems, all authorized systems are assessed for vulnerabilities by an independent entity to ensure that weaknesses are mitigated.
The main sealing system used by IAEA are reported in Table 2.3.
Surveillance is used to detect all movements of nuclear material and spent fuel containers and to confirm that containment is maintained, information regarding locations and material quantities are correct, and that IAEA devices are not tampered with. Thanks to it, IAEA is able to ensure the absence of undeclared operations and continuously monitor a specific activity for a short period of time. Surveillance refers to both human and instrumental one. However, human observation cannot be done every single day continuously, so IAEA developed a set of optical surveillance system that can provide effective surveillance when inspectors are not physically available.
Optical surveillance is also used to identify items during unattended NDA measurements and indications of tampering on the instruments in use. A surveillance needs a camera's

Seals	**Description**
Cap seal	Cap seal applied to a big variety of containments. It is verified at the IAEA headquarters after being removed.
Improved adhesive seal	Commercial sealing tape. The seal must be destroyed to be removed.
Secure vial sealing container	Plastic container for DA sample vials. It is verified at IAEA headquarters after removal.
Fibre optic general purpose seal	Fibre optic seal verifiable in situ.
Ultrasonic sealing bolt	Bolt seal mainly used under water to seal the lids of spent fuel assemblies containers.
Electronic optical sealing system	Reusable seal made of a fibre optic loop and electronic seal. Laser pulses monitor the loop, every opening and closing of the seal is stored in the seal. It is verified by a dedicated reader.

Table 2.3. Main IAEA sealing systems [10]

field of view to cover the entire area of safeguards interest in order to be effective and must be able to capture any movement of the safeguarded items. Besides, the picture-taking interval is set such that its direction of movement can be determined. The image recording frequency may be set at a fixed time interval which is significantly shorter than the fastest removal time, or may be triggered by scene change detection or other external triggers. Some of these surveillance systems can also transfer data to IAEA Headquarters or to an IAEA regional office automatically.

Among the surveillance equipment, single cameras are used for easy and somewhat difficult access areas, instead, multi-camera are useful for larger and more complex facilities. Short term surveillance is done in activities including open core monitoring, while surveillance is done for remote monitoring.

Today's technology mainly relies on Digital Image Surveillance (DIS) systems because they offer several advantages like the reduction of moving parts, a much higher reliability than other systems like film and videotape technologies, an improved authentication and encryption, and a more facilitated remote monitoring.

The main optical surveillance systems and subsystems used by IAEA are reported in Table 2.4 [10].

Surveillance system	**Description**
HAWK-SG digital imaging surveillance system	Small, light, battery-powered single camera for easy to access areas and portable applications and short term temporary inspections.
Digital single camera optical surveillance system	Single camera for difficult to access areas.
Next generation of surveillance system	Modular and scalable optical surveillance system.
Digital multi-camera optical surveillance system	Multi-camera surveillance system for up to 16 cameras with remote monitoring capability.
Server based digital image surveillance	Multi-camera surveillance system for up to 6 cameras with remote monitoring capability.
FAST company surveillance system	Multi-camera surveillance system developed by Euratom for joint applications.
High intensity LED light	Battery-powered, modular high intensity light source to back up external light sources for in-air and underwater surveillance applications.

Table 2.4. Main IAEA optical surveillance systems and subsystems [10]

2.2.3 Environmental sampling

Environmental sampling has proven to be an invaluable tool and one of the strongest technical verification measures introduced for IAEA safeguards, it provides a powerful means for detecting undeclared nuclear material and activities and aids the IAEA in drawing credible safeguards conclusions. [12]
Environmental sampling is the collection of environmental samples at a nuclear site combined with ultrasensitive techniques like particle analysis and it is very useful because it can reveal signature of past and present activities in areas where nuclear material is handled. It focuses on the collection of swipe samples inside enrichment plants, in installations with hot cells and in those types of facilities connected with activities under an additional protocol. [10]
Environmental sampling has many advantages with respect to other sample media:

- Samples from inside process buildings give the highest probability of detection for undeclared activities carried out there.
- The use of certified clean swipe media virtually eliminates the background (especially of U) which may dilute or obscure the anthropogenic nuclear signatures present.
- Samples are small, lightweight so they can be shipped and stored easily.
- Samples are well suited to analysis techniques which give the maximum amount of

useful information about the nuclear materials and activities present in the sampled location. [13]

In order to collect the samples, the inspectors use a special sampling kit which is shown in Figure 2.1.

Figure 2.1. IAEA environmental sampling kit. [11]

The IAEA has established a network of analytical laboratories for environmental sample analysis that provides a combination of analytical techniques. In particular, at the IAEA Class-100 Nuclear Material Laboratory (NML) at the Safeguards Analytical Laboratory (SAL) in Seibersdorf, Austria, the environmental samples are received and are coded to keep confidentiality and then are measured by low background γ ray spectrometry to detect the presence of actinide elements or activation products: the samples are provided a γ ray spectrum in the energy range from 5 keV to 3 MeV and if sufficient activity is detected, an evaluation of the spectral peaks can be performed to estimate the activity in the sample of individual γ emitting isotopes.
After that, samples are measured by a X ray fluorescence spectrometry to detect nanogram to milligram amounts of uranium or other relevant elements: the sample is held by a robot arm and irradiated with X rays from an X ray tube, resulting in the emission of fluorescent X rays from elements present on the swipe. These fluorescent X rays are detected using a 100 mm^2 Si(Li) detector placed near the sample. Counting is performed for four or five hours and then the spectra are evaluated to determine the amount of the element presence and also its spatial distribution. This method is completely non-invasive since the subsample can be measured inside its plastic bagging.
α and β counting could also be applied to detect actinides or β emitting isotopes: a gridded ionization chamber counting system can be used to screen radioactive swipe samples for the presence of α or β emitting isotopes. The swipes are subsampled with an adhesive carbon disc, which is placed in the counting chamber and measured for one hour.
The laboratories in the IAEA Network of Analytical Laboratories (NWAL) are designated by the various Member States and must have a quality assurance system which is audited

by the IAEA. Their measurement schemes can be divided into bulk and particle analysis methods. In bulk analysis implies the entire subsample is dissolved and measured. Particle analysis involves the chemical or isotopic measurement of individual micrometer-sized particles containing U or Pu. [10]

2.2.4 Inspections

On-site inspections are crucial for the IAEA safeguards: the Agency has the right and the responsibility to send inspectors in order to verify if there has been compliance with the State declarations. The inspectors have the role of obtaining and verifying on the spot the accounting for materials under safeguards and must report any non compliance.
IAEA is required to to give at least one week's notice of each inspection, indicting the name of the inspectors, the places and times of their arrival and departure, and the materials and facilities to be inspected. Inspectors can be accompanied by representatives of the State but they must not delay or hinder the inspectors in their task. Moreover, it can happen that inspectors are not able to bring all the appropriate equipment, so they must be provided by the State, if request, with all the necessary instruments. Besides, inspectors must be provided with suitable transport and accomodation, and their activities must be set such to ensure the effective discharge of their functions and the minimum inconvenience and disturbance to the State and the facility. Agency inspectors are to be accorded access at all times to all places, data and to any person who deals with materials, equipment, or facilities to be safeguarded. [14]
There are four kind of inspections:

- Ad hoc: they are done to verify the State's initial report of nuclear material and international transfers, and they are used until a facility attachment is in force.

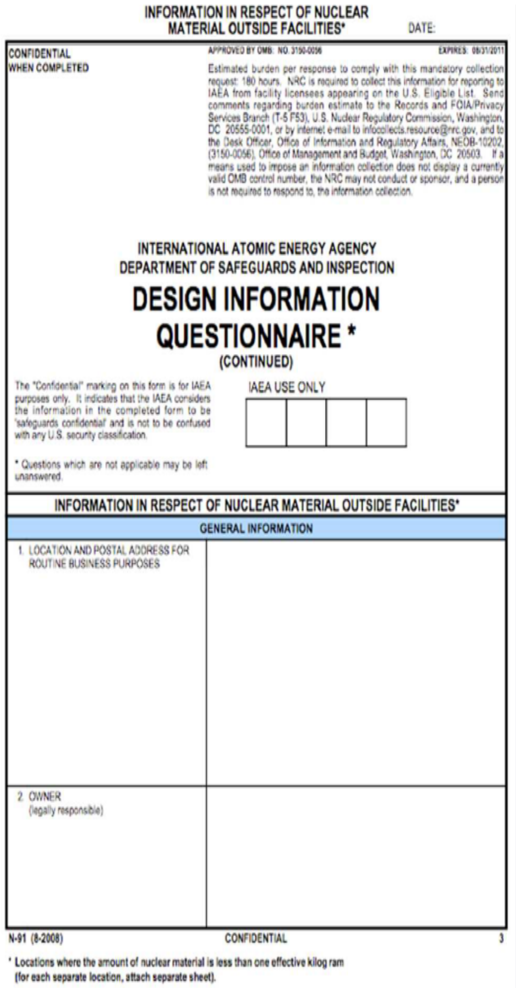
- Routine: these inspections are carried out according to a defined schedule or on short notice and they are performed into those locations within a nuclear facility or other locations containing nuclear material, through which nuclear material is expected to flow.

- Special inspections: they are made to specific locations that address IAEA concerns: the Agency may carry out such inspections if it considers that information provided by the State is not adequate for the Agency to fulfil its responsibilities under the safeguards agreement.

- Safeguards visits: they are made to declared facilities at appropriate times during the lifecycle for verifying the safeguards relevant design information. For example, such visits may be carried out during construction to determine the completeness of the declared design information, or during routine facility operations and following maintenance to confirm that no modification was made that would allow unreported activities to take place, or during a facility decommissioning to confirm that sensitive equipment was rendered unusable. [15] [16]

2.2.5 Verification of information

Another critical aspect of IAEA safeguards is the verification that a facility design and construction (including upgrades and modifications) are not used to further a State's nuclear weapons ambitions. The State must provide relevant nuclear facility design and operating information to the IAEA and this provides the opportunity for the IAEA to verify the safeguards relevant features of the facility and to periodically ensure that those features have not changed. The design information is initially conveyed from the facility operator through the national authorities to the IAEA using the Design Information Questionnaire (DIQ), which is shown in Figure 2.2. Inspectors perform the Design Information Examination (DIE) of declared information, in order to design a safeguards approach for each facility, and the Design Information Verification (DIV) using this information, together with other available information, to confirm that a facility is built and operated as declared.



INFORMATION IN RESPECT OF NUCLEAR
MATERIAL OUTSIDE FACILITIES*

DATE:

CONFIDENTIAL
WHEN COMPLETED

APPROVED BY OMB: NO. 3150-0056

EXPIRES: 08/31/2011

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INTERNATIONAL ATOMIC ENERGY AGENCY
DEPARTMENT OF SAFEGUARDS AND INSPECTION

DESIGN INFORMATION
QUESTIONNAIRE *
(CONTINUED)

The "Confidential" marking on this form is for IAEA purposes only. It indicates that the IAEA considers the information in the completed form to be 'safeguards confidential' and is not to be confused with any U.S. security classification.

* Questions which are not applicable may be left unanswered.

IAEA USE ONLY

INFORMATION IN RESPECT OF NUCLEAR MATERIAL OUTSIDE FACILITIES*

GENERAL INFORMATION

1. LOCATION AND POSTAL ADDRESS FOR ROUTINE BUSINESS PURPOSES	
2. OWNER (legally responsible)	

N-91 (8-2008) CONFIDENTIAL 3

* Locations where the amount of nuclear material is less than one effective kilog ram (for each separate location, attach separate sheet).

Figure 2.2. An example of a DIQ. [17]

To verify the above aspects of a facility, the inspectors compare the information in the DIQ with what they find during the inspection. They will also look up articles and information on the facility to confirm the location and exterior appearance. Then at the facility, they will use sophisticated tools to verify that the buildings match the construction drawings and facility plans. Additionally, the inspectors must verify the material flow paths (entry and exit points) and the equipment placed at key measurement points. They must take note of hallways, connected piping and any changes or alterations to the facility relevant to drawing a conclusion about the safeguards of the facility.
The current DIV activities are very labor intensive, and the individual experience and capabilities of the inspector play an important role. The DIV capability of the IAEA inspectors would benefit from tools able to assist in managing the complex facility information and improving their ability to extract the relevant data like the following ones:

- 3-Dimensional Laser Rangefinder: the 3-Dimensional Laser Rangefinder (3DLR) is a laser-based survey tool that has been adapted for use in facility DIV inspections. The system is able to create 3D maps of rooms and objects and to identify changes in positions and modifications with a precision on the order of millimetres (Figure 2.3).

- Compton Gamma Radiation Imaging: 3DLR images can only show what is already visible to the inspector and not materials that could be hidden piping or diverted in undeclared flow pathways. Adding a radiation detection component would enhance the ability to meet this need, so a Compton Compact Imager (CCI) has been developed, able to provide an image with a high spatial resolution showing the location of radiation sources.

- Virtual Reality for Facility Models: the complexity of the information is a tough task for the inspectors. Projected virtual reality offers the ability to integrate this collection of information into a tool that improves the inspector's ability to prepare for and perform the DIV inspection.

- Robotics: robots can enter areas the person cannot (for example, high radiation fields or contaminated areas), they have been successfully employed in other fields to map areas and carry sensors and they can produce facility maps to compare against declarations, both inside and outside the facility.

- Ground Penetrating Radar: Ground Penetrating Radar (GPR) uses pulses of electromagnetic radiation in the microwave band of the radio spectrum and reads the reflected signal to detect subsurface structures and objects without drilling, probing or otherwise breaking the ground surface. [18] [19]

DIV is performed throughout the life cycle of a facility and its frequency depends on technology sensitivity, operating capacity and operational status and schedule, but it is usually done at least once per year. [17]

Figure 2.3. A portable 3D laser scanning system. [20]

2.2.6 Safeguards agreements

In 1970, the Board of Governors established a Safeguards Committee which developed a document entitled "The Structure and Content of Agreements between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons" (INFCIRC/153 (Corr.)), which was approved in 1972 and used as the basis for negotiating safeguards agreements under the NPT. A model agreement based on INFCIRC/153 (Corr.) was eventually developed and published in 1974 and agreements concluded on the basis of it are commonly referred to as Comprehensive Safeguards Agreements (CSAs).

Once a comprehensive safeguards agreement enters into force, the State is required to submit to the IAEA an initial report of all nuclear material in the State, in accordance with the terms of the agreement. The IAEA then verifies the initial report with the purpose of ensuring that the declaration is correct and complete. The State is also required to provide with a list of all of its nuclear facilities, as also defined in the agreement, and information on their design. This list must include all facilities, not only the operating ones, even if they contain no nuclear material or are under construction. The IAEA then verifies the design information to ensure that the facility is constructed and is operated as declared by the State. [21]

Measures in the CSAs include:

- IAEA collection of environmental samples in facilities and at locations where inspectors have access during inspections and design information verification.

- IAEA use of unattended and remote monitoring of movements of declared nuclear material in facilities and the transmission of authenticated and encrypted safeguards-relevant data to the Agency.

- IAEA expanded use of unannounced inspections within the scheduled routine inspection regime.
- IAEA enhanced evaluation of information from a State's declarations, IAEA verification activities and a wide range of open sources.
- State provision of design information on new facilities and on changes in existing facilities as soon as the State authorities decide to construct, authorize construction or modify a facility. The IAEA has the continuing right to verify the design information over the facility's lifecycle, including decommissioning.
- Closer co-operation between the IAEA and the State (and regional) systems for accounting for and control of nuclear material in Member States.
- Provision of enhanced training for IAEA inspectors and safeguards staff and for Member State personnel responsible for safeguards implementation. [16]

In conjunction with a CSA, a Small Quantity Protocol (SQA) can be concluded. It is meant for States with little or no nuclear material subject to safeguards and with only limited nuclear activities, with the purpose of minimizing the burden of safeguards activities. [22]
INFCIRC/153 (Corr.) also provided the framework for the Voluntary Offer Agreements (VOAs) of the five NWSs: NNWSs must implement safeguards to everything regarding nuclear which can result into slower and more expensive works, while NWSs are not required to conclude a safeguards agreement, so they voluntarily conclude the VOAs to allay concerns that safeguards could lead to commercial disadvantages. Though, VOAs involve only civil nuclear in the five NWSs, their military facilities are still not under safeguards agreements. [17]
However, these kinds of agreements focus on only declared material at declared facilities, they assume a State declares everything. The discoveries in Iraq of a clandestine nuclear weapons programme in the early 1990s emphasized the increasing importance of assurances regarding the absence of undeclared nuclear material and activities in States committed by treaty to non-proliferation. Thus, it was necessary to update the safeguards system by adding measures giving the Agency improved capabilities to detect clandestine nuclear activities. The IAEA Secretariat's response, with the strong support of Member States, was an extensive multi-year programme (Programme 93 + 2) to improve the effectiveness and efficiency of the safeguards system. One objective was to establish the technical and legal basis through which safeguards, while continuing to provide assurance regarding the correctness of States' nuclear material declarations, could also address their completeness. This effort culminated in 1997 with the Board of Governors approving the Model Protocol Additional to Safeguards Agreements, the Additional Protocol (AP), and published as INFCIRC/540 (Corr.). [23]
Measures in the APs include:

- State provision of information about, and IAEA inspector access to, all parts of a State's nuclear fuel cycle (including uranium mines, fuel fabrication and enrichment plants, and nuclear waste sites) as well as to any other location where nuclear material is or may be present.

- State provision of information on, and IAEA short-notice access to, all buildings on a nuclear site.
- IAEA collection of environmental samples at locations beyond declared locations when deemed necessary by the Agency.
- IAEA right to make use of internationally established communications systems, including satellite systems and other forms of telecommunication.
- State acceptance of IAEA inspector designations and issuance of multiple entry visas (valid for at least one year) for inspectors.
- State provision of information about, and IAEA verification mechanisms for, its research and development activities related to its nuclear fuel cycle.
- State provision of information on the manufacture and export of sensitive nuclear-related technologies, and IAEA verification mechanisms for manufacturing and import locations in the State. [16]

After an AP has entered into force, the State is required to submit an initial declaration of the information required in the protocol, and thereafter to submit updates as provided for in the protocol. An AP is not a stand-alone document, it can only be concluded in conjunction with a safeguards agreement, although the two instruments need not be concluded simultaneously. [21]
As of 1^{st} June 2021, APs are in force with 137 States and Euratom, while other 14 States have signed an AP but is not into force yet. [24]
If a State is found to be in non-compliance, the IAEA Director General shall report to the IAEA Board of Governors. Examples of violation of safeguards agreement are the diversion of nuclear material from declared activities, obstruction of IAEA inspectors activities and interference with the operation of safeguards equipment.
If a non-compliance case occurs, a multi-stage process is applied by IAEA which can be summed up into seven steps:

1. Inspectors report any non-compliance to the Director General who transmits the report to the Broad of Governors.
2. The Broad of Governors calls upon the involved State to remedy immediately any non-compliance.
3. The Broad of Governors reports the non compliance to all members and the Security Council and General Assembly of the United Nations.
4. After considering the reports transmitted by the Director General, the Security Council has three possibilities: it does nothing, it limits its reaction to a Presidential statement, or it adopts a specific resolution.
5. Upon request of the IAEA the Security Council could adopt a specific resolution with the purpose of temporarily increasing IAEA access rights to locations, facilities, individuals, equipment and documents.

6. If the Director General is not satisfied, the Security Council could then adopt a second specific resolution requiring the non-compliant State to immediately suspend all sensitive nuclear fuel cycle related activities, and requesting the Director General to report within 60 days on whether the State has complied.

7. If the State does not comply with the previous two resolutions, the Security Council could adopt a third resolution requiring all States to immediately suspend all military cooperation with the non-compliant State. [25]

2.3 Nuclear decommissioning

The term decommissioning encompasses all the management and technical actions associated with ceasing operation of a nuclear installation and its subsequent dismantling to facilitate its removal from regulatory control (delicensing). These actions involve decontamination of structures and components, dismantling of components and demolition of buildings, remediation of any contaminated soil and removal of the resulting waste. Decommissioning activities are intended to place the facility in a condition that provides for the health and safety of the general public and the environment while, at the same time, protecting the health and safety of the decommissioning workers. [26]

2.3.1 Reasons and strategies

There is a wide variety of reasons why a nuclear facility has to be permanently shut down.

- Uneconomical operation: the operating costs are too high for the owner's resources.
- Technical obsolescence: closure for this reason might arise if another facility offered the same or wider applications more effectively or in case of termination of the design life of major components.
- Conclusion of research programmes: nuclear facilities can be built to support one or more major research programmes and if the operators are not able to market the specialized services once the research programmes are concluded, closure may be necessary.
- Safety consideration: the closure of the facility could occur if the regulatory body required safety improvements to conform with modern standards and these improvements were too expensive to be implemented.
- Change in government policy: a government may decide that the facility is no longer required to support national interests or priorities (as occurred in Germany and in Italy).
- Accident: the facility may have to be closed because of an accident or an unplanned event resulting in extensive contamination or structural damage. [27]

Three decommissioning strategies have been defined: immediate dismantling, deferred dismantling and entombment.

Immediate dismantling (DECON) starts shortly after shut down and all components and structures that are radioactive are cleaned or dismantled, packaged and transported to a low-level waste disposal site (if available) or stored temporarily on site. Once this task is completed, the facility can be used for another power plant or other purposes, without restrictions.
Deferred dismantling (SAFSTOR) is a strategy in which the nuclear plant is kept intact and placed in protective storage for a very long time (up to 60 years), and afterwards it is dismantled. This method, which involves locking that part of the plant containing radioactive materials and monitoring it with an on-site security force, uses time as a decontaminating agent. Once radioactivity has decayed to low levels, the activity is the same as in DECON. All building structures and systems which are necessary for workers and public safety shall be maintained in service during the safe storage period. A pre-condition to reach the safe storage condition is that the fuel has been removed from the plant and that radioactive liquids have been drained from systems and components and then processed.
Entombment is a strategy where the radioactive inventory is enclosed in a monolithic structure, for example concrete, to secure the public safety. The monolithic structure should ensure integrity for about 100 years to derive benefit from the decay of the nuclides. After the entombment period, all enclosed components are very low radioactive and the assumption should be that dismantling at that time can be performed in a conventional way. During entombment the plant remains under a nuclear license. This approach is usually used only in some specific cases like after an accident but it is not usually recommended. [28]
Many actors come into play in the decision of the strategy to adopt.

- The national policy and the regulatory framework.
- Proposed reuse of the facility or site and the desired end state.
- The physical and radiological status of the facility.
- Safety and nuclear security aspects.
- The environmental impact of the facility and of its decommissioning.
- Societal and economic factors and the socioeconomic impact of decommissioning.
- The availability of infrastructure for radioactive waste management.
- The availability of financial resources for decommissioning.

This process of decision of the decommissioning strategy is outlined in Figure 2.4 [29].

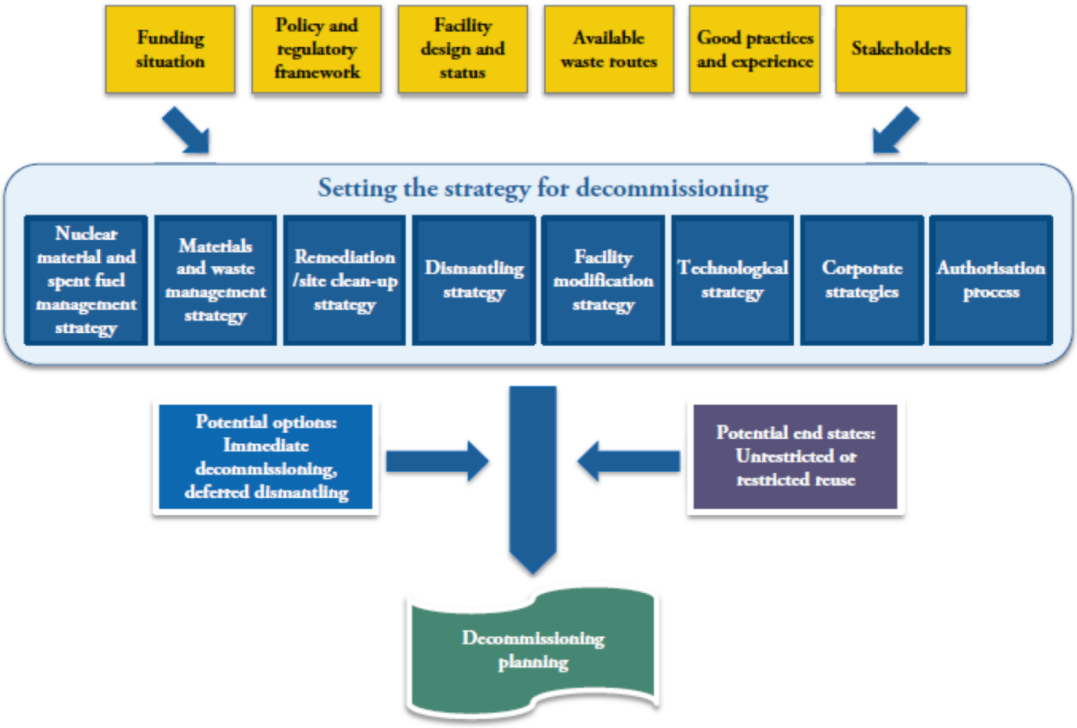

Figure 2.4. How to choose the decommissioning strategy. [29]

2.3.2 Responsibilities of the parties involved

The national government must establish and maintain a governmental, legal and regulatory framework where all aspects of decommissioning can be planned and carried out safely. This framework includes a clear allocation of responsibilities and requirements with respect of financial assurance for decommissioning.
The regulatory body must regulate all aspects of decommissioning throughout all stages of the facility's lifetime, from initial planning for decommissioning during the siting and design of the facility, to the completion of decommissioning actions and the termination of authorization for decommissioning. Other important responsibilities of the regulatory body are the safety requirements for decommissioning, including those for radioactive waste management,and taking actions to ensure that the regulatory requirements are met.
The licensee must plan for decommissioning and conduct the decommissioning actions in compliance with the authorization for decommissioning and with requirements derived from the national legal and regulatory framework. The licensee is responsible for all aspects of safety, radiation protection and protection of the environment during decommissioning.
Responsibilities with respect to financial provisions for decommissioning shall be set out in national legislation. Adequate financial resources to cover the costs associated with safe decommissioning must be available when necessary and the mechanism used to provide them must be consistent with the cost estimate for the facility and shall be changed if necessary.

It is important to update this cost estimate for decommissioning on the basis of the periodic update of the initial decommissioning plan or on the basis of the final decommissioning plan. If this financial assurance for the decommissioning of an existing facility has not yet been obtained, adequate financial resources must be put in place as soon as possible. If a sudden shutdown of the facility occurs, provisions must be put in place to enable use of the financial resources for decommissioning when they are needed. If the decommissioned facility will be released with restrictions on its future use, financial assurances must be such that financial resources are available for monitoring, surveillance and control of the facility throughout the necessary time period. [30] Financing methods vary from country to country but the following three strategies are the most common ones. Prepayment is the method where money is deposited in a separate account to cover decommissioning costs even before the plant begins operation. This may be done in a number of ways but the funds cannot be withdrawn other than for decommissioning purposes. In external sinking fund the capital is built up over the years from a percentage of the electricity rates charged to consumers. Proceeds are placed in a trust fund outside the utility's control. The last one foresees a surety fund, letter of credit, or insurance purchased by the utility to guarantee that decommissioning costs will be covered even if the utility defaults. [2] Based on the latest information provided by EU Member States, European nuclear operators estimated in December 2014 that € 263 billion will be needed for nuclear decommissioning and radioactive waste management until 2050, with € 123 billion for decommissioning and € 140 billion for spent fuel and radioactive waste management as well as deep geological disposal. [31]

2.3.3 Decommissioning stages

Plant characterization is the first stage of decommissioning and it is the description and inventory of the nuclear facility to be decommissioned. It is a very important process that will affect all subsequent stages. The aim is determine the nature and extent of radiological contamination, and collect any information available on all the elements that make up the plant in order to support the evaluation of remediation technologies and the management in decommissioning. Plant characterization can be defined in seven steps:

1. Historical assessment: it consists of the investigation to collect existing information describing a plant complete history from the start of activities to the present time with the purpose of identifying potential or known sources of contamination or activation, and differentiating impacted areas from non-impacted areas.

2. Implementation of calculation methods: it is the use of computer codes to provide values of the neutron induced activity in the nuclear reactor. One important part of this step is to decide whether the theoretical calculations are sufficient for the subsequent planning of the decommissioning activities or whether they should be supplemented by a more or less detailed sampling and measurement plan. In this context, historical data may play an important role.

3. Site reconnaissance: it consists of the facility walkthroughs to gather information concerning the physical and radiological conditions in the plant. It is made of the structural survey, which is an in situ physical characterisation of equipment, components and structures of the plant, hazardous materials survey, which localizes,

identifies and quantifies hazardous materials, and scoping radiometric survey, which provides preliminary radiological information and hazard assessment and consists of judgement measurements based on the historical site assessment data.

4. Non-destructive radiological analyses: it is the qualitative and quantitative survey and assessment of contamination by the so called easy-to-measure radionuclides (γ emitting ones) present in the plant. Firstly, a sampling and analysis plan is defined by means of a proper statistical approach based on the type, quantity and quality of the data, then the measures are performed and finally there is the reviewing and evaluation of the results.

5. Destructive radiological analyses: it is the precise information on radiological conditions with respect to the so called difficult-to-measure radionuclides (α, β, X-ray or weak γ emitting ones) present in the plant. In this case, after having defined a plan for the analyses, sampling is performed and the resulting samples are pre-treated and shipped to the laboratory for the analyses.

6. Data and sample management: all data are recorded and kept in a computer database.

7. Definition of Homogeneous Groups: a homogeneous group is made of components with uniform radiological characteristics, with the same ratio of α, β and γ contamination. All three types of radioactivity are measured and correlation between γ and the other two can be determined. All components are assigned to homogeneous groups and must not be mixed up subsequently for the entire duration of the process. [32]

The second stage is related to engineering and licensing: as with any industrial project, engineering is present all along the decommissioning process lifetime. The project is planned in detail, from the actual dismantling, through waste management, to the full clean up of the site turning into the required end state. The plan defines exactly what has to be removed, how and in which order, and then safety analysis is performed and protection systems are defined. Everything is planned paying attention to safety, environmental protection, quality, costs and human resources. Once the plans are set and documented, the proposal is submitted to various authorities, both local and national, namely for environmental protection and nuclear safety.
Once plan has been approved, it is time for the third stage, which is the actual dismantling. The facility is being prepared for the work to be done step by step, zone by zone. Enclosing the perimeter is the first task, then all the residual liquids are drained from tanks and pipes, the electricity is disconnected and an autonomous power system is set up, local ventilation systems are installed, and if necessary concrete, or even lead walls are erected to protect workers from radiation, finally storage areas are created for radioactive waste and for materials that are not contaminated which must undergo exit checks. During the dismantling phase, besides paying attention to homogeneous groups there is another golden rule: the components must be removed in the order from the cleanest to the most contaminated. About 90% of the material can be defined as clearable, uncontaminated. Waste traceability and its physical and radiological characterization are fundamental. Each drum and container, once filled and sealed, is catalogued and labelled, while the content is described, photographed and measured. A γ spectrometer is used to measure the radiation coming from inside the drums and it is able to precisely measure all γ radiation. Thanks to

the homogeneous groups established during the plant characterization, α and β radiations can be calculated by measuring only the γ radiation.
As far as decontamination is concerned, chemical solutions are generally most effective on non-porous surfaces. The choice of decontamination agents is based upon the chemistry of the contaminant, the chemistry of the substrate and the ability to manage the waste generated during the process. Strong mineral acids (like nitric acid, sulphuric acid and phosphoric acid) are used to attack and dissolve metal oxide films and lower the pH of solutions in order to increase solubility or ion exchange of metal ions. Foam is used on its own or as a carrier for chemical decontamination agents. This process is well developed and widely used, especially for large components with complex shapes or large volumes. Chemicals gels are used as carriers of chemical decontamination agents and are sprayed or brushed onto a component or surface, allowed to work, then scrubbed, wiped, rinsed or peeled off. Mechanical decontamination methods can be used on any surface where contamination is limited to near surface material. Dusting, vacuuming, wiping and scrubbing involve the physical removal of dust, aerosols and particles from building and equipment surfaces using common cleaning techniques. Abrasive cleaning uses an abrasive medium such as plastic, glass or steel beads, or grit such as garnet, soda or aluminium oxide. It is used to remove smearable or fixed contamination from metal surfaces such as structural steel components and hand tools and also from concrete surfaces and coatings. High pressure water processes use a pressurized water jet to remove contamination from the surface of the workpiece, the contamination being removed by the force of the jet.
As far as dismantling is concerned, in mechanical cut techniques the direct action of the tool on the workpiece produces a cut. This is achieved by the tool fracturing, cleaving or eroding the workpiece surface. Most of them produce easily handled secondary waste streams which can be collected by local extraction systems. They also produce much fewer airborne fumes than thermal techniques, thus simplifying viewing of the cutting operation, although cutting speeds are generally lower. Shears can be manually, pneumatically, hydraulically or electrically actuated and are used for segmenting metal and crushing concrete. Sawing techniques make use of shearing processes, normally produced when a hard cutting edge bears against a softer material which is to be separated. Different kinds of mechanical sawing techniques can be used throughout decommissioning operations for different purposes. Mechanical sawing machines range in size from small hand-held hacksaws to very large and heavy bandsaws capable of cutting steam generators. There are three main mechanical saw types: reciprocating saws (including hacksaws and guillotine saws), bandsaws and circular saws.
In Figure 2.5 it is shown an example of a reciprocating saw in use.
Cutting with thermal techniques is done by melting or burning the material by use of a high concentration of energy in the processing area. This creates a certain amount of aerosol particles, dust and fumes, which require suitable precautions such as suction units with highly efficient filter systems to protect workers and avoid spreading of contamination. The flame cutting process uses a torch where a heating gas reacts with oxygen to form a heating flame. This flame heats up the material to a temperature where burning of the material can begin. Plasma arc cutting is based on an electrical arc between an electrode inside a torch and the workpiece. A gas (such as argon, nitrogen, hydrogen or air) is injected into the arc inside the torch, turning it into a plasma with a temperature of more than 10,000 °C. The plasma gas exits the torch through a nozzle as a jet with

Figure 2.5. Reciprocating saw. [33]

high kinetic energy and capable of melting not only every metal, but also blowing away the molten material. Moving the torch creates a kerf with a clean cutting edge. Cutting is also possible under water but performance decreases with increasing water depth. [33] [34]
The characterized drums are sent to areas for the fourth stage, which is waste management. Waste management covers all technical and administrative activities involved in the handling, sorting, characterization, treatment, conditioning, storage, transportation and disposal of radioactive waste from a nuclear installation. It is the phase where the mass and the volume of radioactive waste are reduced as much as possible and after that, final waste packages are created in processes called waste conditioning and they are stored temporarily. All drums are directed to dedicated facilities depending on the type of treatment they require. It can be divided into four steps: pretreatment, treatment, conditioning and disposal. Figure 2.6 shows a scheme of the various phases involved.
Pretreatment is considered as any operations prior to treatment and includes operations such as waste collection, segregation, chemical adjustment and decontamination, and is performed to reduce the amount of waste needing further treatment and conditioning, storage and disposal, to adjust the characteristics of the waste, to make the waste more amenable to further processing, and to reduce or eliminate certain hazards posed by the waste owing to its radiological, physical and chemical properties.
The treatment phase includes reduction in the volume of the waste, removal of radionuclides, change of the form or composition or properties of the waste. Compaction is a suitable method for reducing the volume of certain types of waste: a drum full of soft, compactable waste is put into a hydraulic press and compressed. This reduces its volume by up to 70%. Then there is metal melting, with which mass of metal waste can be reduced by 90%. During the melting process, all radioactive contamination is separated from the metal which can be then recycled. Sludge and the filters that collect all the contaminants are stored in new drums which have to be characterized again. Incineration of combustible solid waste normally achieves the highest reduction in volume as well as yielding a stable waste form: some flammable waste is burned in special incinerators reducing its mass by 70%. All the radioactivity is not lost but it is concentrated in the ashes and filters which

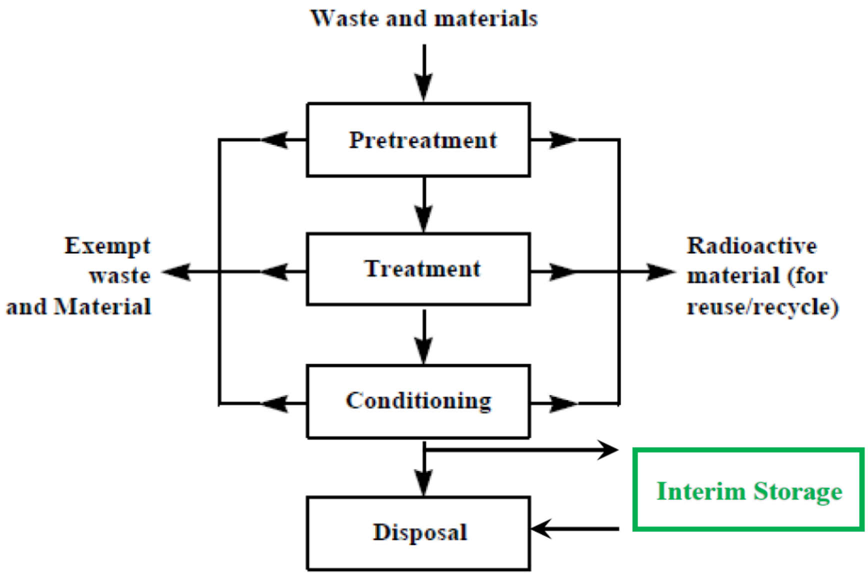

Figure 2.6. Stages of radioactive waste management. [35]

are collected and confined in new drums which also need to be characterized. Special filters separate radionuclides from clean water which can be released into the environment after the final check, while all radioactivity is concentrated in radioactive sludge which is collected into drums.
Waste conditioning of radioactive waste consists of operations that produce a waste package suitable for safe handling, transport, storage and disposal. Waste packages produced by conditioning should satisfy the respective acceptance criteria, which are the radiological, mechanical, physical, chemical and biological characteristics of waste packages that will compliant with the storage or disposal facility safety case. Therefore, the regulatory body and organizations operating or planning to operate transport services, storage facilities and disposal facilities should be consulted in deciding which types of pretreatment, treatment and conditioning will be necessary. The conditioned waste must be monolithic, homogeneous and must have low contaminant leaching, good stability and good mechanical strength. Cementation is a widely used technique available even in mobile units: in this process, the waste, water and additives are dosed from a tank into the drum at the waste loading station. Then, the waste-loaded drum is transferred to the in-drum mixer, where cement is added and mixed with the waste. It is applicable to solid and a variety of wet wastes (50% of water) and the volume of cemented waste is typically twofold in comparison with the original waste volume. Since it is not allowed the disposal of liquids, as it is not safe, liquid waste is often converted into a solid form by solidifying it in a suitable matrix, such as cement, bitumen or polymer for low level and intermediate level waste, or glass for high level waste, in accordance with the waste acceptance criteria. Overall, liquid treatment is very effective and it reduces the volume of the waste by 99%. Now that the mass and the volume of radioactive waste is concentrated and reduced, all post

treatment products are encapsulated in large sealed containers. To immobilize the waste inside these containers, a special cement is poured into them to fully encapsulate the waste in the middle. These special containers filled with radioactive waste and cement are called final waste packages and they are designed to be buried underground. They are then put into the final disposal facility if present or, if not, into an interim storage facility waiting for the final disposal. [35] [36] [37]
Figure 2.7 shows an example of a drum undergoing cementation.

Figure 2.7. Cemented drum. [38]

As final step, the licensee has to prepare a final decommissioning report to demonstrate that the end state of the facility as specified in the approved final decommissioning plan has been reached. This report shall be submitted to the regulatory body for review and approval. The regulatory body reviews the final decommissioning report and evaluates the end state to ensure that all regulatory requirements and end state criteria, as specified in the final decommissioning plan and in the authorization for decommissioning, have been met. On the basis of this review and evaluation, the regulatory body decides on the termination of the authorization for decommissioning and on the release of the facility or the site from regulatory control. If the approved decommissioning end state is release from regulatory control with restrictions on the future use of the remaining structures, appropriate controls and programmes for monitoring and surveillance are established and maintained for the optimization of protection and safety, and protection of the environment. These controls must be approved by the regulatory body and responsibility for implementing and maintaining these controls and programmes must be clearly assigned. If radioactive waste is stored on the site after that the decommissioning has been completed, a revised or new separate authorization for the waste storage facility shall be sought from the regulatory body. This authorization includes requirements for the decommissioning of the storage facility. Inputs from the public must be addressed before authorization for decommissioning is terminated.

2.3.4 Radioactive waste classification

Radioactive waste includes any material that is either intrinsically radioactive, or has been contaminated by radioactivity, and that is deemed to have no further use. It can be classified according to different ways.

- By half-life: short lived waste, long lived waste.
- By activity concentration: low level waste, intermediate level waste, high level waste.
- By physical state: solid, liquid, gaseous.
- By origin: Nuclear fuel cycle, isotope production, etc.

There is no one universal approach but the IAEA has made a qualitative classification to make a link between waste characteristics and disposal method and, on the basis of it, every State has made its own quantitative national classification. The IAEA ended up with six categories of radioactive waste, as shown in Figure 2.8.

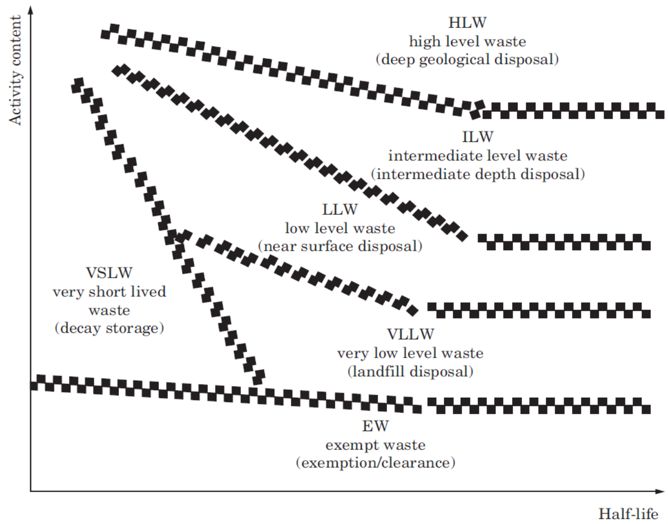

Figure 2.8. The waste classification scheme suggested by IAEA. [39]

Exempted Waste (EW) is waste that meets the criteria for clearance, exemption or exclusion from regulatory control for radiation protection purposes.
Very Short Lived Waste (VSLW) is waste that can be stored for decay over a limited period of up to a few years and subsequently cleared from regulatory control according to arrangements approved by the regulatory body, for uncontrolled disposal, use or discharge. It includes waste containing primarily radionuclides with very short half lives often used for research and medical purposes.
Very Low Level Waste (VLLW) is waste that does not necessarily meet the criteria of EW, but that does not need a high level of containment and isolation and, therefore, is suitable for disposal in near surface landfill type facilities with limited regulatory control. This

class includes soil and rubble with low levels of activity concentration.
Low Level Waste (LLW) is waste that is above the clearance level and with limited amounts of long lived radionuclides. It requires robust isolation and containment for periods up to a few hundred years. Suitable for disposal in engineered near surface facilities typically from the surface to around 30 m depth. It may include short lived radionuclides at higher levels of activity concentration and long lived radionuclides at relatively low levels of activity concentration. LLW comprises some 90% of the volume but only 1% of the radioactivity of all radioactive waste.
Intermediate Level Waste (ILW) is waste that contents long lived radionuclides, so it requires a greater degree of containment and isolation than that provided by near surface disposal facilities. Disposal is at greater depth than that of near surface disposal facilities, for example at the order of tens of metres to a few hundred of meters (intermediate depth disposal). ILW typically comprises resins, chemical sludges, and metal fuel cladding, as well as contaminated materials from reactor decommissioning.
High Level Waste (HLW) is waste with high levels of activity concentrations, heat generation by radioactive decay or with large amounts of long lived radionuclides. Disposal is in deep, stable geological formations of several hundred meters depth or more below the surface. There are two distinct kinds of HLW: used fuel that has been designated as waste and separated waste from reprocessing of used fuel. HLW accounts for just 3% of the volume, but 95% of the total radioactivity of produced waste. [40]

2.3.5 Radioactive waste disposal

Disposal is the final step in the management of radioactive waste. It aims to provide safety through emplacement of waste in facilities designed for appropriate levels of containment and isolation. Such facilities are designed and maintained to encompass both natural and engineered barriers for adequate radiation protection of people and environment over long periods of time. A number of disposal options have been developed for final management of radioactive waste. The options vary in the amount and characteristics of different waste types, the specifics of national legislation and geological differences. [41]
LLW and short-lived ILW (up to 30 years) are typically sent to land-based disposal immediately after its packaging, which are called near-surface disposal. This means that for the majority of all of the waste types produced by nuclear technologies, a satisfactory disposal means has been developed and is being implemented around the world. There are two kinds of near-surface disposal. The first ones are near-surface disposal facilities at ground level which are on or below the surface. Waste containers are placed in constructed vaults and when full the vaults are backfilled. They will be finally covered and capped with an impermeable membrane and topsoil. These facilities may incorporate some form of drainage and a gas venting system. Examples include:

- Centre de l'Aube in France (Figure 2.9).
- El Cabril LLW and ILW disposal facility in Spain.
- LLW Repository at Drigg in Cumbria in the UK.
- LLW Disposal Center at Rokkasho-Mura in Japan.

Figure 2.9. Center de l'Aube facility in France. [42]

The last ones are near-surface disposal facilities in caverns below ground level for which underground excavation of caverns is required. These facilities are at a depth of several tens of metres below the surface and accessed through a drift. These facilities will be affected by long-term climate changes and this effect must be taken into account when considering safety, as such changes could disrupt these facilities. An example is the underground repository at Olkiluoto in Finland for LLW and ILW.
As far as HLW is concerned, the first step is storage to allow decay of radioactivity and heat, making handling much safer. Storage may be in ponds or dry casks, and beyond it, many options have been investigated which seek to provide publicly acceptable, safe, and environmentally sound solutions to the final management of radioactive waste. The most widely favoured solution is deep geological disposal. The long timescales over which some waste remains radioactive has led to the idea of deep disposal in underground repositories in stable geological formations. Isolation is provided by a combination of engineered and natural barriers (rock, salt, clay) and no obligation to actively maintain the facility is passed on to future generations. The waste packaging, the engineered repository, and the geology all provide barriers to prevent the radionuclides from reaching humans and the environment. In addition, deep groundwater is generally devoid of oxygen, minimising the possibility of chemical mobilization of waste. As a general rule, three broad types of geology found to be suitable. The first ones are higher strength rocks, which typically comprise crystalline igneous, metamorphic rocks or geologically older sedimentary rocks, where any fluid movement is predominantly through discontinuities/faults (granite for example). The second ones are lower strength sedimentary rocks typically, comprising geologically younger sedimentary rocks where any fluid movement is predominantly through the rock mass itself (many types of clay or mud rock for example). The last ones are evaporites which typically comprise halite (rock salt) or other evaporites that have resulted from the evaporation of sea bodies containing dissolved salts. Deep geological disposal is the preferred option for nuclear waste management in most countries but the only purpose-built deep geological repository that is currently licensed for disposal of nuclear material is the Waste Isolation Pilot Plant (WIPP) in the USA (Figure 2.10), but it does not have a licence for disposal of used fuel or HLW. Plans for disposal of spent fuel are particularly well advanced in Finland, Sweden, and France. In Canada and the UK, deep disposal has been selected and the site selection processes have started.

Figure 2.10. The Waste Isolation Pilot Plant in the USA. [43]

Another proposed deep geological disposal concept is for a mined repository comprising tunnels or caverns into which packaged waste would be placed. In some cases, the waste containers are then surrounded by a material such as cement or clay (usually bentonite) to provide another barrier (called buffer and/or backfill). The choice of waste container materials and design, as well as the buffer/backfill material, varies depending on the type of waste to be contained and the nature of the host rock-type available. The contents of the repository would be retrievable in the short term, and if desired, longer-term. The Swedish proposed KBS-3 disposal concept (Figure 2.11) uses a copper container with a steel insert to contain the spent fuel. After placement in the repository about 500 metres deep in the bedrock, the container would be surrounded by a bentonite clay buffer to provide a very high level of containment of the radioactivity in the spent fuel over a very long time period.

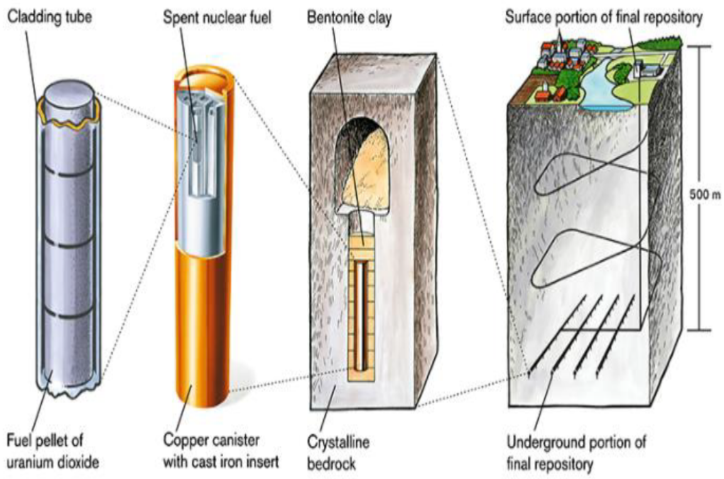

Figure 2.11. The KBS-3 concept for final disposal. [44]

Deep borehole disposal has been also considered as an option for geological isolation. The concept consists of drilling a borehole into basement rock to a depth of up to about 5000 metres, emplacing waste canisters containing used nuclear fuel or vitrified radioactive waste from reprocessing in the lower 2000 metres of the borehole, and sealing the upper 3000 metres of the borehole with materials such as bentonite, asphalt or concrete. The disposal zone of a single borehole could thus contain 400 steel canisters each 5 metres long and one-third to half a metre in diameter. The waste containers would be separated from each other by a layer of bentonite or cement and the contents would not be retrievable. Deep borehole concepts have been developed (but not implemented) in several countries, including Denmark, Sweden, Switzerland, and the USA. Compared with deep geological disposal in a mined underground repository, placement in deep boreholes is considered to be more expensive for large volumes of waste. This option was abandoned in countries such as Sweden, Finland, and the USA, largely on economic grounds. The borehole concept remains an attractive proposition for the disposal of smaller waste forms including sealed radioactive sources from medical and industrial applications. [45] [46]

Chapter 3

Safeguards applied during decommissioning

3.1 Spent fuel safeguards

Irradiated nuclear fuel is included in the range of nuclear materials under international safeguards agreements because of its fissile material content [47] because it keeps being a potential source of uranium and plutonium for use in war. Thus, measurement techniques must be deployed to avoid any non-peaceful scenario. There are two main types of techniques, which were already introduced in the first chapter: DA techniques and NDA techniques.

3.1.1 Destructive Analysis techniques

DA is a measurement technique such that the sample to be measured is not returned to the initial material batch it was taken from. It is used to verify that diversion of nuclear materials has not occurred, to provide assurance of the quality and independence of on-site measurements, and to carry out periodic verification of operator measurement systems. The samples size in DA is usually small and they need to be conditioned. The advantages are that DA has high accuracy and direct traceability but it is time consuming and it can create secondary waste. The steps of DA measurements are:

1. Taking of independent samples.
2. Conditioning of these samples at the facility to ensure that they maintain their chemical form and their integrity during transport.
3. Packaging, sealing and shipment of samples to the IAEA NML at the SAL in Seibersdorf where the samples are measured and analysed through DA techniques to verify the uranium and plutonium content.
4. Statistical evaluation of analysis results. [10]

DA measurements for element or isotope assay may be based on physical or chemical principles. Chemical methods allow to determine the amount of U or Pu contained in a sample.

These methods rely on characteristic chemical properties of the individual element. For example, in classical chemical TITRATION methods, a reagent is added to change the oxidation state of the element under investigation and then, by measuring the amount of reagent, the amount of U or Pu in the sample can be obtained. Instead, in gravimetric analysis the compound under investigation is brought to a defined stoichiometry so, by measuring the mass change induced to the material, the element content in the initial compound can be determined. Physical methods are based on the radiation emitted or absorbed by the material under investigation. If the radiation originates from the electron shell (X ray) then it is characteristic for the element. If the radiation originates from the nucleus, it is characteristic for the nuclide, hence the isotope. Radiation of both types may either be emitted spontaneously, like in decay, or induced (due to irradiation). Mass spectrometry can be applied for the determination of the isotopic composition of the element under investigation. NDA techniques are usually used for spent fuel measurements but there are also some DA techniques that came up to be useful. [48]
Isotopic Dilution Mass Spectrometry (IDMS) is the basic technique for safeguards verification measurements of uranium or plutonium in all samples of spent fuel. IDMS method works as follows: fixed amount of the isotopic spike (the solution with certified isotopic composition in sharp contrast with the isotopic composition of the sample analysed) is added to the known aliquot of the analysed solution with predetermined isotopic composition of the element measured. After the operation of isotopic exchange, measurement is performed of the isotopic ratio of the base isotope (isotope of the element analysed with highest content) to the added spike isotope. [49] For low burnup spent fuel, a spike of $^{242}_{94}Pu$ or $^{244}_{94}Pu$ is used. The chemical treatment of spent fuel samples is performed using a fully automated robotized system in order to reduce radiation dose levels and improve work efficiency. The resulting uranium and plutonium fractions are then evaporated to dryness and redissolved in nitric acid to yield solutions containing about 1 mg of uranium and 50 ng of plutonium per microlitre. The isotopic ratios of both the spiked and unspiked aliquots are measured by Thermal Ionization Mass Spectrometry (TIMS) and the uranium and plutonium contents are calculated accordingly. [10]
The TIMS (Figure 3.1) is a magnetic sector mass spectrometer that is capable of making very precise measurements of isotope ratios of elements that can be ionized thermally, usually by passing a current through a thin metal ribbon or ribbons under vacuum. The ions created on the ribbons are accelerated across an electrical potential gradient and focused into a beam via a series of slits and electrostatically charged plates. This ion beam then passes through a magnetic field and the original ion beam is dispersed into separate beams on the basis of their mass to charge ratio. These mass-resolved beams are then directed into collectors where the ion beam is converted into voltage. Comparison of voltages corresponding to individual ion beams yield precise isotope ratios. [50]

Figure 3.1. A TIMS. [10]

3.1.2 Non Destructive Analysis techniques

NDA measurements do not produce significant changes to the item physical or chemical properties. NDA instruments range in size and complexity from small portable units used by safeguards inspectors during on-site verification activities to large in situ NDA systems designed for continuous unattended in-plant use.
NDA techniques mainly rely on the measurements of two kind of radiations: γ rays and neutrons.
γ rays are measured through γ ray spectrometry: every γ ray emitting radionuclide emits γ rays with well defined energies so by reconstructing the spectrum of the γ rays emitted the element can be uniquely recognized. To detect γ rays, the radiation must interact with a detector to give up all or part of the photon energy and the consequent liberated electrical charges are used to produce a voltage pulse whose amplitude is proportional to the energy deposited by a γ ray in the detector. These pulses are then sorted according to amplitude (energy) and counted using appropriate electronics.[10]
Neutrons are emitted from non-irradiated nuclear fuel mainly in three ways: by spontaneous fission of uranium and plutonium, by induced fission by neutrons from other sources, and by α particle induced reactions (α, n) involving low atomic number elements such as oxygen and fluorine. Neutrons have mass but no electrical charge. This means that they cannot directly produce ionization in a detector, and therefore cannot be directly detected. Thus, in neutron detectors an incident neutron interacts with a nucleus to produce a secondary charged particle and these charged particles are then directly detected and from them the presence of neutrons is deduced. The most common reaction used for high efficiency thermal neutron detection is:

$$ {}^{1}_{0}n \ + \ {}^{3}_{2}He \ \rightarrow \ {}^{0}_{1}p \ + \ {}^{3}_{1}H \ + \ 0.765 \ MeV $$

where the proton and the tritium are detected by a gas filled proportional counters using ${}^{3}_{2}He$ fill gas.

It is not possible to use energy discrimination to distinguish neutrons from different sources like in γ rays case, therefore, traditional spectroscopy techniques cannot be used. However, there is a characteristic time distribution difference between (α, n) neutrons and neutrons from a fission event that can be exploited: fissions will produce two or three neutrons simultaneously, while (α, n) neutrons are produced individually and randomly. This allows coincidence counting techniques to be used to distinguish the prompt fission neutrons from the random (α, n) neutrons. The die-away time is defined as the characteristic time a neutron will survive before it is absorbed in the ^3_2He tubes or escapes the counter. Gross neutron counting, instead is simply the sum of all neutrons, without any coincidence technique. Thus, the neutron source cannot be characterized, but a significant number of neutrons can sometimes be enough to indicate the presence of fissile nuclear material. The even isotopes of plutonium ($^{238}_{94}$Pu, $^{240}_{94}$Pu and $^{242}_{94}$Pu) undergo spontaneous fissions, so the neutrons from them are detected and counted. Since no external neutron source is required to induce fission, assay systems of this type are known as passive neutron counters. $^{235}_{92}$U, $^{238}_{92}$U, and $^{239}_{94}$Pu do not spontaneously fission at a high enough rate to allow passive assay techniques to be used. Therefore, an external neutron source is used in order to induce fission in the sample. Assay systems using this technique are known as active neutron counters. [51]
Spent fuel verification can be performed through many techniques that can rely on γ rays, neutrons or a combination of both of them.
An established technology used for spent fuel safeguards measurements is the Fork detector, which is based on both γ rays and neutrons. The Fork instrument has gained widespread because of its reliability, portability, speed of measurement, and simplicity. However, careful evaluation of Fork measurement data is crucial due to the sensitivity of the measured signals to initial enrichment, burnup, cooling time and fissile content. The Fork detector has two arms, each of which contains an ion chamber for γ flux measurement and two $^{235}_{92}$U fission chambers for neutron measurement. In the IAEA Fork design, all detectors are embedded in a polyethylene block. The Fork detector head is mounted on a stainless steel pipe through which the connecting cables are fed to the electronic unit at pond side. Separate detector heads are used to measure Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) type fuels (Figure 3.2). Measurements are generally performed in water-filled ponds prior to dry storage cask loading, with both the detectors and fuel assembly under water. To perform a measurement, the irradiated fuel assembly is lifted by the operator's crane and moved into position between the tines of the Fork detector. Interactive software guides the user through the measurement procedure and simultaneously collects neutron and γ ray data. The ratio of the neutron to γ ray data is used to characterize a particular type of fuel assembly, giving information related to its neutron exposure in the reactor, its initial fissile fuel content and its irradiation history. The Fork detector is also used by the European Atomic Energy Community (Euratom) but in their case in each arm the ion chamber and one of the fission chambers are also covered with a cadmium liner (while the other fission chamber is bare). [47]
The Safeguards MOX PYthon (SMOPY) system is based on the combination of passive gross neutron counting and low resolution γ spectroscopy and it is able to characterize any type of spent fuel. The measurement data are evaluated through online interpolation tools. SMOPY device is an integrated system which is composed of the following items.

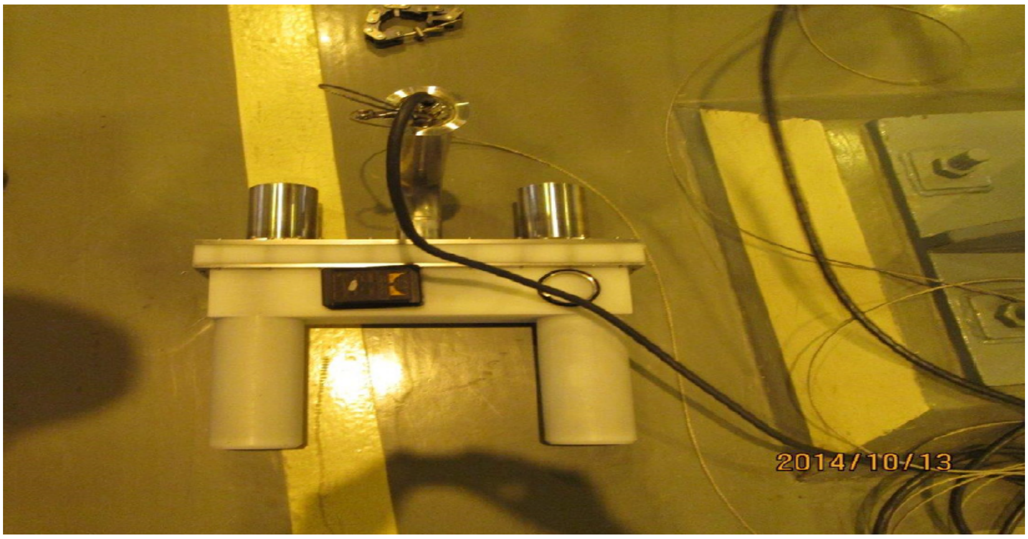

Figure 3.2. PWR Fork detector head assembly showing the two measurement arms. [47]

- The cylindrical measurement head with one high efficiency fission chamber for neutron counting and a micro room temperature γ spectrometric probe. γ rays are detected by a collimated CdZnTe detector which is also protected by a tungsten shielding with remote control that automatically controls the count rate.
- The carrier which holds the measurement head. The carrier bottom fits the racks for accurate positioning and its top fits operator's fuel moving tool.
- The portable electronic cabinet which includes both neutron and γ electronic cards.
- The laptop PC which gets inspectors data, controls the measurement, gets measured values, interprets them and immediately provides the inspector with worthwhile information for appropriate field decisions.

The device is placed over the storage hole of the spent fuel assembly which is then lifted through the open measurement cavity, and either it can be scanned or selected parts can be measured. (Figure 3.3)
The main feature of SMOPY is its capability of distinguishing Mixed OXide fuel (MOX) from Low Enriched Uranium (LEU). It uses passive neutron counting alone in two cases: if the neutron emission (NE) value is less than 10^8, the assembly is surely high burnup Uranium OXide (UOX), if the NE value is greater than $5*10^9$, the assembly is surely MOX. For NE values between 10^8 and $5*10^9$, a reductio ad absurdum method implemented in the SMOPY device is used.
SMOPY can be used also for Partial Defect Tests (PDTs) and so to verify if some rods have been removed. The methodology performed is the following one.

1. Use declared data to perform parametric depletion online calculations.
2. Build the correlation the correlation of NE as a function of burnup and of the ratio ${}^{134}_{55}Cs/{}^{137}_{55}Cs$ again as function of burnup.

3. Perform a γ spectrometry to get the measured $^{134}_{55}Cs/^{137}_{55}Cs$ ratio.

4. Check the declared irradiation history and burnup using measured and calculated $^{134}_{55}Cs/^{137}_{55}Cs$ ratio.

5. Extract NE from depletion calculation.

6. Perform a passive neutron counting.

7. Correct the multiplication effects regards to boron concentration and verified burnup.

8. Do a consistency check between calculated and measured NE: if actual measured NE does not fit the theoretical NE within calculation and measurement uncertainty, the fuel cannot be declared non defective. [52]

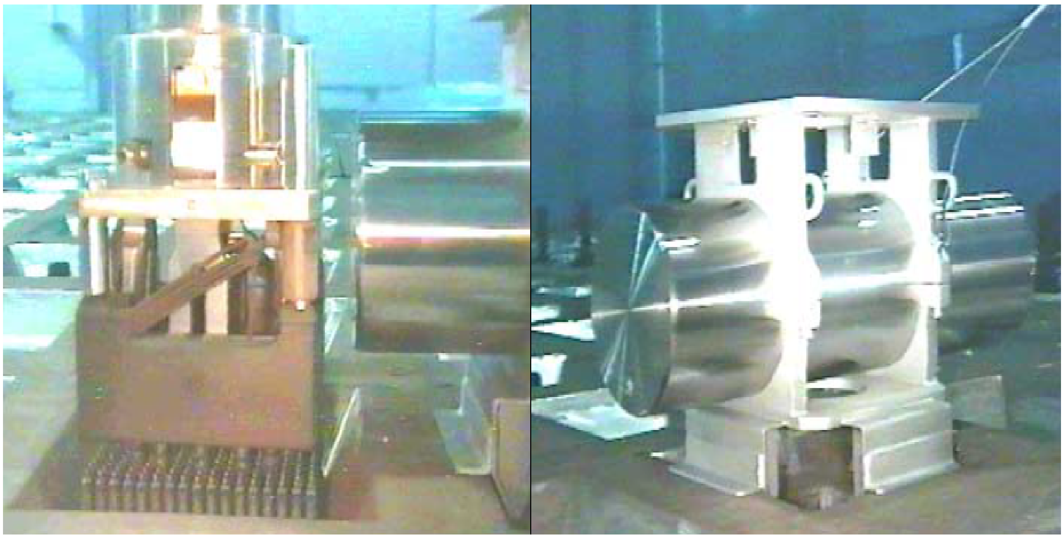

Figure 3.3. SMOPY device under water during measurement [52]

The Digital Cherenkov Viewing Device (DCVD) is an image intensifier viewing device that is sensitive to ultraviolet radiation in the water surrounding spent fuel assemblies. Cherenkov light is a faint emission accompanying the passage of charged particles through a transparent medium at speeds faster than the speed of light in that medium. Since high-energy γ rays can generate high-velocity electrons by Compton scattering, Cherenkov light can be emitted as a result of γ radiation fields as well. [53]The digital Cherenkov viewing device is based on an ultra-sensitive camera detecting ultraviolet light. The camera is connected to a computer that uses specialized software to analyse the image (Figure 3.4). Its specialized lens and sensor capture ultraviolet light emitted from spent fuel assemblies, and the light reveal key details about their characteristics. This is used to verify spent fuel ponds, ensuring that spent fuel was not diverted and substituted with a non-fuel assembly. Furthermore, this device does not get immersed in the fuel pond, so it does not get contaminated with radioactive elements. [54]

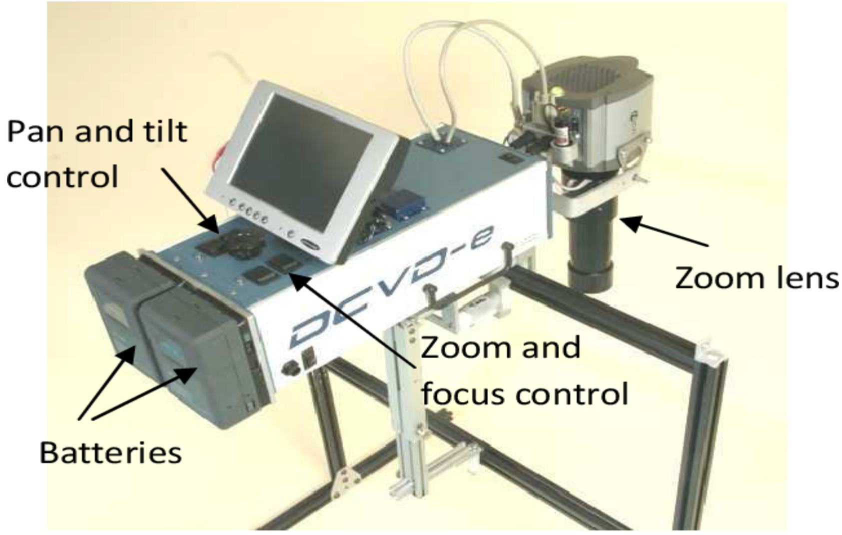

Figure 3.4. The mounted DCDV. [55]

The Spent Fuel Attribute Tester (SFAT) consists of multichannel analyzer electronics unit and a NaI or CdZnTe detector. The SFAT detector and its lead shielding are housed in a watertight stainless steel container which is submerged in a storage pond: the SFAT is used for taking measurements from the top of a fuel assembly as it sits in the storage rack. A watertight collimator pipe is attached below the detector housing to permit only radiation from the principal assembly rather than from adjacent assemblies to reach the detector. The SFAT provides a qualitative verification of the presence of spent fuel through detection of γ rays of $^{137}_{55}$Cs (662 keV) for fuel that has cooled for longer than four years or of $^{95}_{40}$Zr/$^{95}_{41}$Nb (757/766 keV) for fuel with short cooling times. Activation products such as $^{60}_{27}$Co are also identifiable. The SFAT is particularly helpful in situations when Cherenkov radiation is weak (for example, when the spent fuel has low burnup and/or a long cooling time, or when the water in the storage pond is insufficiently clear) therefore Cherenkov viewing cannot provide verification.

The IRradiated fuel Attribute Tester (IRAT) is a small, lightweight CdZnTe based detector that can be suspended from a spent fuel pond bridge and used to differentiate irradiated non-fuel items from irradiated fuel items that are stored in spent fuel storage ponds.

Both the SFAT and the IRAT have the same theoretical concept in respect of using a γ spectrometry that analyses the specific spectrum of the γ ray from the target and most of the components and measurement system are very similar. However, the IRAT uses a smaller detector compared to the SFAT that is well-fitted to the low intensity γ ray measurement. Besides, the IRAT has not a vertical collimator but a horizontal collimator to acquire the signal directly from the target without air collimator pipes so that the item should be lifted up and moved to the side of the equipment. The advantage of the collimator geometry of the IRAT is identifying the middle or low section of the non fuel items where is not able to be verified with the SFAT. However, the signal is subject to saturate by the effect of the gamma ray from highly irradiate metal so that the more complex shielding system is required to prevent the saturation in case of the IRAT. The watertight air collimator pipe system of the SFAT permits only radiation from the target assembly rather than from adjacent assemblies to reach the detector. For these reasons, the effect without air collimator pipes and ringshaped shied around the detector have to be considered to enhance the quality of the conceptual design. [10] [56]

Figure 3.5. An IRAT. [54]

The Neutron and Gamma Attribute Tester (NGAT) is a compact neutron and γ detector which aims to underwater verification of spent fuel containing $^{244}_{96}$Cm, which emits neutrons, and $^{137}_{55}$Cs, which emits γ rays at 662 keV. The neutron detector of the NGAT has $^{3}_{2}$He as gas inside a $^{10}_{5}$B detector or a fission chamber, High Density PolyEthylene (HDPE) as moderator and lead as γ shielding, while the γ detector is a 20 mm^3 crystal CdZnTe detector. [10] [57]
Safeguards evolve and new verification capabilities are developed. The Passive Gamma Emission Tomography (PGET) is currently just a prototype but it is the most promising method to detect replacement of a single fuel rod in a light water fuel spent fuel assembly. The PGET has the outside appearance of a torus with a central hole for the fuel element. It weighs 520 kg in air and 300 kg under water. Each of the two detector arrays contains 104 CdTe detectors embedded in tungsten collimators. The fuel element will be placed in the central tube of 450 mm height and of 325 mm inner diameter. The central tube is large enough to surround PWR 17x17 type of fuel. The large flange has an outside diameter of 955 mm. A circular rail goes around inside the toroidal casing. The two detector banks are positioned onto the rail and can make a slow rotation movement around the center which is controlled by a stepping motor. The nuclear measurement electronics consists of two boards, which are inside the two detector banks (Figure 3.6). Emitted radiation along different directions is detected by the collimated detectors and a section image is calculated from this measured dataset. The image shows a rod–to–rod distribution of the γ emitter concentration. Replacement or missing of rods can be revealed by visual or computer based evaluation of the image. [58]

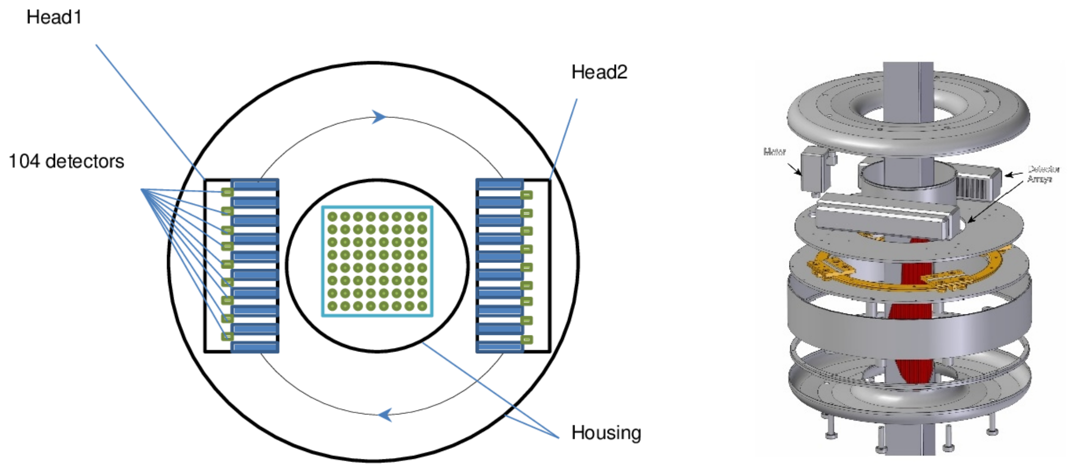

Figure 3.6. Exploded view of PGET and prototype arrangement. [59]

3.2 Safeguarding facilities during decommissioning

IAEA safeguards involve verification activities at a wide range of facilities in a variety of operational phases, including decommissioning. The decommissioning stage involves activities that will ultimately lead to the removal of all nuclear material and other radioactive material from the facility. However, as long as a risk of unauthorized removal of nuclear material or of sabotage leading to unacceptable radiological consequences remains, nuclear security measures should be maintained. The application of these measures should be based on a graded approach, taking account of the category of nuclear material and its potential for sabotage leading to unacceptable radiological consequences. [60] All this process can be subdivided into three parts: when the facility is shutdown (cessation of operations without removal of all nuclear material from the facility), when it is closed-down (operations have been stopped and the nuclear material removed but the facility has not been decommissioned), and when it is decommissioned (residual structures and equipment essential for its use have been removed or rendered inoperable so that it is not used to store and can no longer be used to handle, process or utilize nuclear material).
Shutdown nuclear facilities have an inventory of nuclear material. During this phase, the IAEA applies the same safeguards approach as when the facility was under normal operation to verify that the nuclear material has not been diverted. The IAEA conducts interim inspections to verify nuclear material transfers from the facility. Periodic Physical Inventory Verification (PIV) and DIV activities are still conducted to ensure the continued presence of the declared nuclear material, that the declared facility design is still valid, and that the plant is not being misused for non-peaceful nuclear purposes. The inventory of some nuclear material may become zero during the shutdown period, and so no further verification would be required. States are required to inform the IAEA on any modification of the facility status that can impact safeguards implementation. Many States do not inform the IAEA about the expected time duration of an extended shutdown because this may be unknown when and for which reasons the shutdown occurs. For example, the facility may be shut down for economic or political reasons, such that when the conditions

change, the facility may be restarted. The time and cost to make repairs may also take time to understand, especially as regulations may be changing because of relevant events. After the shutdown the facility switches to the closed-down phase and it is in state of decommissioning. Before the decommissioning starts, States should provide IAEA with a plan about the facility's decommissioning where the time schedule for removing or rendering inoperable the essential equipment of the facility is explained in detail. IAEA legal instruments (the safeguards agreements) do not require the State to provide information about the places where the removed major equipment or components will be transferred and this lack of information could be significant with respect to sensitive equipment that could be used to build an undeclared facility. IAEA verifies the decommissioning process through inspections, DIVs, and Complementary Accesses (CAs), if an AP is in force, throughout the decommissioning phase. Installed IAEA equipment will be removed, as appropriate. Nuclear material recovered from equipment or residing in equipment transferred to waste would be verified during inspections before the nuclear material is transferred from the facility, while material that cannot be removed from equipment may be difficult to measure and an appropriate NDA measurement method would be used. During the decommissioning period, the IAEA assesses periodically the level of effort required to reactivate the facility with the remaining equipment.
Eventually the facility is decommissioned: after that IAEA has confirmed that sufficient residual essential equipment has been removed (so that the facility can no longer be used to store or handle, process, or utilize nuclear material), and that the efforts and resources to reactivate the facility are equal to or exceed that necessary for a new facility, IAEA will confirm to the State that the facility has been 'decommissioned for safeguards purposes' and removed from the list of facilities subject to safeguards. After this time, for countries with an AP in force, the only verification activity that can be performed at the decommissioned facility are CAs to confirm the continued decommissioned status, while no further verification activity is performed in States with only CSAs.
To make an example, the safeguards applied to a Light Water Reactor (LWR) during the three previously explained phases will be shown.
In the shutdown phase the reactor still has fresh and irradiated fuel which may be shipped to other facilities. There are three possible diversion scenarios and one misuse scenario: removal of fresh fuel assembly or pins from storage area, removal of irradiated fuel assembly or pins from reactor core or during transfer from reactor core, removal of irradiated fuel assembly or pins from spent fuel pool or during transfer to cask for storage or shipment, and irradiation of undeclared nuclear material in the reactor core to produce plutonium or ${}^{233}_{92}U$ for recovery elsewhere, respectively. To contrast these scenarios, annual PIV and DIV, and random interim inspections (usually one inspection per year per five reactors) are performed. The DIQ is updated and operating schedule (transfers from core and shipments from facility) and the schedule of shipments are submitted.
During the closed-down stage the reactor has no fresh or irradiated fuel. Decommissioning work is underway with removal or destruction of essential equipment according to the decommissioning plan. There is no diversion scenario, as no declared nuclear material is present. Instead, there are three possible misuse scenarios: replacement of damaged or removed essential equipment, refueling and restart of reactor with undeclared fuel, irradiation of undeclared nuclear material in the reactor core to produce plutonium or ${}^{233}_{92}U$ for recovery elsewhere. The safeguards measures adopted are annual DIV with PIV and

possible CSAs if an AP is present. Additional DIVs may be required to verify removal or destruction of essential equipment. The DIQ is updated and the operating schedule (decommissioning activities and removal/destruction of essential equipment) is submitted. Finally in the decommissioned phase sufficient essential equipment has been removed or disabled to prevent future operation of facility without great effort. Since there is no declared nuclear material is present, like in the previous stage, there are no diversion scenarios. Misuse scenarios are also not regarded as credible. Only possible CAs for AP-States are possible and no information are submitted, as the reactor is no longer considered a nuclear facility for safeguards purposes. [61]

Chapter 4

Italian situation

Italy has been in the past among the leading countries in the pacific use of nuclear energy, but, as a consequence of the 1987 referendum, it decided to shutdown all operating power plants and to leave uncompleted the plants under construction. These plants need thus to be decommissioned in a safe and secure way.

4.1 Italian nuclear agencies

Even though there are no nuclear plants under construction, there are some institutions in Italy that deal with nuclear energy, also concerning decommissioning and safeguards aspects.

4.1.1 ENEA

The Agenzia nazionale per le nuove tecnologie, l'energia e lo sviluppo economico sostenibile (ENEA) is an agency governed by public law which carries on its activities in accordance with the directives of the Comitato Interministeriale per la Programmazione Economica (CIPE). It is placed under the authority of the Ministry for Economic Development, and consults the Ministry of University and of Research regarding research and development programmes, the Ministry for the Ecological Transition in relation to projects which could affect the natural environment and the Minister for Foreign Affairs with regard to international activities (Decree n° 36/99).
As regards new technologies, energy and the environment, the ENEA has some duties. It must carry out and promote studies and research on technological aspects and on the effects of the development and use of technologies, including the economic and social consequences, and on the safety of nuclear installations and protection against ionizing radiation. It cooperates in the scientific, technical and industrial fields, with the international bodies involved in the same sectors. It also formulates and implements plans for the dissemination of know-how and research results to government departments. Finally, it gives opinions and carry out technical controls on behalf of government, regional and local authorities.
To reach these objectives, the ENEA may conclude agreements with the competent Ministries, the regions, the autonomous provinces of Trento and Bolzano, local authorities and

certain local firms, as provided for by Act n° 142 of 8th June 1990. [62]

4.1.2 SOGIN

Legislative Decree n° 79 of 16th March 1999, adopted in implementation of Directive 96/92/EC of the Parliament and of the Council of 19th December 1996 concerning common rules for the internal market in electricity, established a new company named Società per la Gestione degli Impianti Nucleari (SOGIN). It is responsible for the decommissioning of Italian nuclear plants and for the management of radioactive waste, including those produced by industrial, research and nuclear medicine activities. Entirely owned by the Ministry of Economy and Finance, the SOGIN operates according to the strategic guidelines of the Italian government.
Following its creation, the SOGIN set up a consortium with the company Fabbricazioni Nucleari and the ENEA in order to ensure the planning and coordination of the dismantling of research installations belonging to the ENEA and which are associated with the nuclear fuel cycle. Following this, the consortium was terminated and in 2003 SOGIN was charged with operating ENEA's fuel cycle research reactors and the Bosco Marengo plant, acquired in 2005 from the company Fabbricazioni Nucleari. [62]
The SOGIN Group company Nucleco is the leading company in Italy in the field of radiological services, in the management of radioactive waste and in the decontamination and remediation of nuclear plants and industrial sites. [63]
The SOGIN also has the task of locating, designing, building and managing the Deposito Nazionale, a surface environmental infrastructure, where all radioactive waste can be safely stored. The Deposito Nazionale will be flanked by the Parco Tecnologico: a research center, open to international collaborations, in the field of decommissioning and radioactive waste management.
The SOGIN also coordinates the activities set out in the agreement signed between the Italian government and the Russian Federation as part of the Global Partnership program. In particular, the agreement concerns the dismantling of Russian nuclear submarines and the management of radioactive waste and irradiated fuel. In addition to this important intergovernmental collaboration, the company has always been involved at an international level, with two foreign offices in Moscow and Bratislava, on three areas.

- Development of relationships and collaborations with international organizations and foreign operators, public and private, to favor the exchange of know-how applicable to the decommissioning of Italian nuclear plants.

- Commercial development with the acquisition of projects, studies, consultancy and technical services on the dismantling of plants, the management of radioactive waste, as well as on safety and radiation protection.

- Support to Italian institutions to fulfill the provisions of international treaties and commitments. [64]

4.1.3 ISIN

The Ispettorato nazionale per la sicurezza nucleare e la radioprotezione (ISIN) is the independent regulatory authority responsible for nuclear safety and radiation protection, in accordance with the Directives 2009/71/Euratom and 2011/70/ Euratom.
It carries out the investigations related to the authorization processes, the technical assessments, the control and supervision of nuclear installations no longer in operation and in decommissioning, of research reactors, plants and activities related to the management of radioactive waste and spent nuclear fuel, of nuclear materials, of passive physical protection of nuclear materials and installations, of the activities of use of ionizing radiation sources and of transport of radioactive materials. ISIN issues the certifications required by current legislation on the transport of radioactive materials and the technical guides in matters of competence. Furthermore, the Inspectorate provides support to the competent ministries in the drafting of acts of legislative rank and to the Civil Protection Authorities in the field of planning and response to nuclear and radiological emergencies. It carries out the control activities of environmental radioactivity required by current legislation and it ensures the fulfillment by the Italian State of the obligations deriving from international agreements on safeguards, the representation of the Italian State within the activities carried out by international organizations and the European Union (EU) in matters of competence, and the participation in international and EU processes for assessing the nuclear safety of nuclear plants and in the management of irradiated fuel and radioactive waste in other countries.
ISIN is the National Warning Point and the competent national authority, in accordance with the International Conventions, on prompt notification in case of a nuclear accident and on assistance in the event of a nuclear accident or a radiological emergency as well as in the scope of the European Commission's system for rapid exchange of information in case of a radiological emergency (Decision 87/600/Euratom). Moreover, ISIN represents Italy in the international system for the communication of information on events classified according to the International Nuclear and radiological Events Scale (INES). The Inspectorate is also assigned the inspection functions for compliance with the provisions on nuclear safety and radiation protection, exercised by its inspectors, as Judicial Police Officers (Article 10 of Legislative Decree n° 230/1995 and subsequent amendments).
The ISIN participates, on behalf of Italy, in the activities carried out by international organizations operating in the nuclear sector: the IAEA, the Nuclear Energy Agency at the Organization for Economic Cooperation and Development (OECD), the Western European Nuclear Regulators Association (WENRA), the Heads of the European Radiological Protection Competent Authorities (HERCA), the European Association of Competent Authorities for the Safe Transport of Radioactive Material (EACA). In particular, as far as the collaboration with the IAEA, the ISIN participates in the inspections carried out by the IAEA and it must ensure the compliance of Italy with international treaties on safeguards. The Inspectorate also has the task of transmitting and validating the information for the European Commission to be sent to the IAEA and preparing the annual reports. Italy must grant the IAEA access to any place within a site and to any deactivated plant outside the plants where nuclear materials are used, as well as the locations where it wants to take environmental samples. [65]

4.2 Italian nuclear facilities

The SOGIN is responsible for the decommissioning of the four Italian nuclear power plants of Latina, Caorso, Garigliano and Trino, of the plants linked to the nuclear fuel cycle, which are the Fabbricazioni Nucleari plant in Bosco Marengo, IPU and OPEC in Casaccia, ITREC in Rotondella and Eurex in Saluggia, and of the reactor Ispra-1 in Ispra. [66]
Once completed the work of decommissioning, the waste, already conditioned and stored in the temporary deposits of the site, will be ready for be transferred to the Deposito Nazionale (reaching the brown field).
With the availability of the Deposito Nazionale, radioactive waste will be gradually removed and temporary repositories dismantled. The site will thus be brought back to the state of green field, that is to a condition without radiological constraints that will allow its reuse. [67]

4.2.1 Latina

The Latina nuclear power plant was the first to be built in Italy and it was gas technology reactor, a GCR-Magnox. It was shut down in 1987 in the immediate aftermath the referendum. In 1999 the SOGIN acquired it with the purpose of reaching the brown field in 2027. [68]
After the shutdown of the plant, the fuel was removed and the structures were kept safe. The dismantling of fuel loading and unloading machines, which allowed for the replacement and replacement of fuel elements with fresh elements, without turning off the reactor, dates back to the 1990s. The 125,036 irradiated fuel elements of the Latina plant were transferred to England for reprocessing. The residues originating from the reprocessing operations will return to Italy to be temporarily transferred to the Deposito Nazionale. [69]
The overall demolition plan of the reactor building will be carried out in two distinct phases, subject to two different authorization procedures. The first one, authorized by the Ministry of Economic Development on 20^{th} May 2020, provides for a reduction in the volumes of the building, similarly to what is expected on the Windscale (UK) site. The second phase involves the removal of the graphite and the dismantling of its container and the demolition of the entire building. In the meantime, the six lower and upper pipes of the primary circuit, the six blowers and some auxiliary systems and components of the primary circuit were dismantled. During the dismantling, the entire process was kept under control with the aim of minimizing the production of radioactive waste and maximizing the possibility of recovering the materials, also with the adoption of decontamination techniques. Between 2023 and 2025, the structural and plant adaptation of some rooms of the reactor building for temporary storage of radioactive waste is planned. In this way, the temporary storage capacity of radioactive waste on the site will be increased, which is necessary to carry out decommissioning activities without constructing new buildings. The first phase will end in 2027 with the dismantling of the infrastructure and the lowering of the reactor building. With the availability of the Deposito Nazionale, it will be possible to start the second phase of the dismantling of the reactor building which will end with the release of the site without radiological constraints. This condition will be achieved with the dismantling of all the buildings and the transfer of radioactive waste to the Deposito Nazionale,

an operation which will therefore also allow the demolition of the new temporary deposit. [70]
At the end of the 1990s, the first preparatory operation for the dismantling of the turbines was the removal of the lubricating oil and asbestos present in the equipment. Between 2004 and 2006 the turbo-alternators were dismantled and all the materials present were disposed. In 2007, the internal civil reinforced concrete structures were demolished and all equipment removed. In 2013 the building was demolished and cement waste produced was delivered to authorized disposal plants. In the area that housed the turbine building, a reinforced concrete platform was built for the temporary storage of conventional materials deriving from the dismantling activities. [71]
For the temporary storage of radioactive waste produced during the operation of the plant and for those deriving from dismantling activities, between 2009 and 2014 a new temporary deposit was created, consisting of a rectangular concrete building with a surface of about 2,000 m^2. The warehouse, which meets the most modern standards, consists of two storage areas and a service area. The waste storage area is divided into equal parts by a reinforced concrete wall and is served by two overhead cranes operated remotely from a dedicated control room. In 2018, the temporary warehouse went into operation. [72]
The Latina Estrazione COndizionamento (LECO) is the plant for extracting and conditioning the sludge deriving from the previous operation of the treatment systems of radioactive liquid effluents of the Latina power plant in the cement matrix. The LECO facility, built between 2009 and 2017, consists of a plant for extracting sludge from the underground tank, a plant for their conditioning in cement matrix inside cylindrical containers of 440 liters and a tunnel connection between these two plants. Between 2017 and 2018 the plant systems were installed and the functional tests were completed. All operations of extraction, transfer and conditioning of radioactive sludge in the cement matrix will be carried out remotely through a dedicated control room. The artifacts resulting from the operation of the LECO plant will be transferred to the new temporary warehouse. At the end of the operations, the extraction building and the sludge pit will be reclaimed and demolished. [73]
Between 2019 and 2020, some preparatory activities were carried out for the demolition of the six boilers. Among these, the main one is the demolition of the concrete screens, external to the reactor building, of the upper pipes of the primary circuit. Their demolition was started on 4^{th} August 2020. The technique that was adopted is the controlled demolition with cutting in altitude, at a height of about 50 meters by means of a diamond disc, and the subsequent ground handling of the sectioned blocks, of about two tons each, with specially installed tower cranes. Subsequently, the individual blocks were transferred to an area equipped to separate the iron from the concrete. Overall, the works, which ended on 20^{th} October 2020, produced approximately 1,200 tons of material which, after the appropriate checks, will be removed from the site and sent for recovery. The next step is the dismantling of the six boilers, which will lead to a significant reduction in the plant volume, as well as a change in the physiognomy of the reactor building. For the execution of the cuts of the boilers, among the available techniques, cutting with diamond wire has been selected to allow the cutting of components inside the boiler to be carried out at the same time, not preliminarily reachable. The cut will take place from top to bottom through an overhead crane and a cutting machine made ad hoc, able to slide vertically along the boiler. The entire work area will be protected and statically confined from atmospheric

agents with panels and will be kept in a slight depression with respect to the external environment, avoiding dispersion to the outside. After cutting, the produced sections will be transported to the materials treatment facility, to be further segmented and decontaminated. [74]
The construction of a new radioactive wastewater treatment plant is planned, deriving from the decontamination of the components produced by the decommissioning activities and from the washing of the operators' individual protection devices. The plant will be built in an area in front of the building that houses the current active effluent plant, which will be dismantled, to optimize the path of the wastewater discharge pipes. It will consist of three floors and will have an area of 32 x 15.5 meters and a maximum height of 9 meters. The plant will allow, through a combined process of evaporation and mechanical filtration, to treat radioactive waste. The new plant, unlike the previous one, will be located above the ground level and will allow a significant reduction in the production of radioactive waste. In fact, an evaporation system will be adopted which will make it possible to isolate the radioactive part of the wastewater, transforming the conventional part into water vapor. In 2020 the foundation slab and the storage tanks preparatory to construction were completed and the removal of the old line of active effluents began. Between 2022 and 2023, with the entry into operation of the new one, the old plant will be dismantled. [69] [75]
The weakly contaminated metal materials that will be produced by the future dismantling of the active liquid effluent treatment plant, by the reclamation of the reactor building and by the demolition of the boilers, will be treated within a facility currently under construction (materials treatment facility). Inside, the volume reduction of the contaminated metal materials will take place by oxyacetylene cutting and decontaminated with high pressure hydrolaser. In 2020 the civil works and most of the plant works were completed. [69]

4.2.2 Caorso

The Caorso nuclear plant was a 860 MW$_{el}$ BWR. In 1986 it was shut down for the periodic refill of the fuel and because of the 1987 referendum it was never restarted. In 1999 the SOGIN became owner of the plant with the purpose of performing the decommissioning. The decommissioning activities are currently ongoing and it is planned to reach the brown field in 2031. [76]
The first decommissioning activities of the plant concerned the decontamination of the primary circuit. About 2.5 m^3 of LLW were produced, conditioned and safely stored in the site's temporary repositories. The activity, carried out between 2003 and 2004, also reduced the levels of radioactivity in the workplace by more than 200 times. [69]
In the period 2004-2012 the SOGIN removed all the systems and components present in the turbine building (turbo-alternator group, condenser, preheaters, etc.), safeguarding only the systems still required for its reuse, including ventilation, drainage collection and lifting equipment. In particular, the activities began with the removal of all insulation and equipment containing dangerous substances. The first major component to be removed, decontaminated and dismantled between 2004 and 2006 was the turbo-alternator unit (four turbines, one high and three low pressure, and one electric generator). In its place, a materials management station was installed, which came into operation in 2009, which has the necessary equipment for segmentation, radiological control and decontamination of materials. [77]

Between 2007 and 2008 the residual heat removal system building was partially dismantled in its hydraulic part, the cooling towers, about 46 meters long, 31 meters wide and 24 meters high. Overall, 3,100 m^3 of civil works were demolished, producing over 7,000 tons of concrete, and about 300 tons of components were removed. [78]
The exhaust systems and fuel supply pipes of the four emergency diesel generators have been cleaned of asbestos and a new insulating material has been installed which guarantees better performance in terms of safety and resistance to high temperatures. [79]
In December 2020 the construction site of the Waste Route was started, the structure which, by connecting the turbine building, the reactor building and the building auxiliaries, will allow, during the dismantling of the reactor systems, the safe handling of the materials produced. [69]
A project related to the treatment of resins and sludge is currently in progess: it concerns the preparation, transport, treatment and conditioning at the plant of approximately 5,900 drums containing resins and radioactive sludge at the Bohunice plant (Slovakia). In particular, these are 800 tons of spent ion exchange resins and 60 radioactive sludges, both waste produced during the previous operation of the plant. The envisaged treatment process is incineration, while the conditioning of the ashes produced will take place through their insertion into pods, which are in turn incorporated into the cement matrix inside 440-liter stainless steel containers. The objective of the process is to reduce the volume of waste by about 90% (130 m^3 compared to the initial volume of 1,290 m^3), creating final products suitable for transfer to the Deposito Nazionale. At the end, the conditioned artifacts will return to Caorso and will be stored, as LLW, in the site's temporary deposits, ready for their subsequent transfer to the Deposito Nazionale. The overall duration envisaged for the transfer and treatment of the 5,600 radioactive drums and the return of conditioned artifacts is approximately four years (2020-2023). [80]
Most of the radioactivity present in the systems and components of the nuclear power plant is concentrated in the reactor building: the pressure vessel, with the two fuel pools, the suppression pool and all the systems and components associated with them. [81] The 1,032 irradiated fuel elements of the Caorso plant were transferred, between 2007 and 2010, to France, to the reprocessing plant in La Hague. The residues of this operation will return to Italy to be temporarily transferred to the Deposito Nazionale. [69]
During 2020, the commission for the dismantling of the reactor building systems and components was also initiated. The project for the dismantling of the reactor building first involves the progressive dismantling of systems and components present in the different areas and then the dismantling of vessels and internals, divided into three main phases.

1. Preparatory and preliminary activities for the opening of the vessel which provide for structural checks of the pools and the recharging plan, changes to the cooling and water filtration systems and opening of the primary container and reclamation of the heat shield.

2. Opening of the vessel and disassembly, under the water head, of the removable internals with their subsequent storage in the fuel pool.

3. Dismantling operations inside the vessel with removal of the fixed internals, their subsequent cutting and packaging.

This process is expected to last until 2030. [81]

The auxiliary building needs also to be dismantled: it is, together with the reactor and turbine buildings, one of the main complexes of the Caorso nuclear power plant. It is divided into a classified area, which is radiologically controlled, and an uncontrolled area, where there is no radioactivity. In the classified area there are, in addition to the rooms for controlled access, the radioactive waste treatment systems, both liquid and solid, a part of the water purification systems of the fuel pool and condensate treatment. In the uncontrolled area there are the control room, the monitoring system of the gaseous discharges, the ventilation systems and the normal and emergency electrical systems. The dismantling of the auxiliary building systems is scheduled in parallel with the work for the removal of the vessel and internals. It is expected to last nine years (2021-2030). [82]

4.2.3 Garigliano

The Garigliano nuclear plant was a 160 MW_{el} BWR which worked until 1978 when it was shut down for maintenance. In 1982, after the 1980 Irpinia earthquake, it was definitively shut down and in 1999 the SOGIN became its owner and in charge of its decommissioning. The brown field is expected to be reached in 2026. [83]
The first operations after the shutdown of the plant, which took place in 1982, were the drainage of the reactor's hydraulic circuits, the removal of the fuel elements and the emptying of the pool in which they were contained. [84] The 523 fuel elements were removed in several stages from the plant. With these transactions, concluded in 1987, the most of the items were sent to England for reprocessing, while the others were transferred to the Avogadro deposit in Saluggia and are intended for reprocessing in France. [69]
In the years 1996-1998 the radioactive resins used to treat the water of the thermal cycle of the Garigliano plant were conditioned during the operation of the plant. The first operations carried out by the SOGIN concerned the asbestos remediation of the reactor and turbine. Built between 2008 and 2010, they produced waste containing asbestos, which was treated through supercompaction and stored in the plant temporary deposits. Between 2016 and 2020, the restoration activities of the auxiliary plants of the reactor building were carried out in preparation for the subsequent dismantling activities. The reactor vessel will be dismantled following five stages:

1. Removal of materials present in the reactor channel and in the upper part of the vessel housing.
2. Restoration work on equipment, coatings and the flooding system of the vessel, the reactor channel.
3. Opening of the vessel and removal of contaminated materials and equipment used during operation, present in the upper part of the vessel head.
4. Removal and dismantling of the lower internals under the water head.
5. Dismantling of the vessel and the primary circuit. [84]

At the end of 2018, the dismantling, cutting, removal and recycling of the materials of the rotor and the stator of the turbine alternator were completed. Subsequently, the alternator was removed, after having removed the asbestos present and cut it. Dismantling of the

turbine is expected to be completed by 2021. In this regard, in 2020 a cutting machine for large metal components and a sandblaster for decontamination operations have been installed. [69] [85]
The drainage of the floors of the classified areas, of the laundries of the radiation protection devices and the activities foreseen by the surveillance procedures produce radioactive liquid waste that must be suitably controlled and treated to be discharged in compliance with the limits imposed by nuclear standards according to the authorized discharge formula by the decree of decommissioning of the plant. To replace the system operating during the operation of the plant, a new radioactive liquid effluent treatment system has been designed and will be built. The new liquid waste treatment system provides for a drying and evaporation mechanism which allows to minimize the production of secondary waste deriving from the purification process of radioactive liquids. It also produces a high volume reduction of the primary radioactive waste. The drying/evaporation system together with the new storage tanks for the wastewater to be treated and to be discharged after the evaporation process, were built and installed in the specially renovated premises of the plant, while the preparation of the auxiliary systems and of the control systems of the aforementioned plant. Once the installation and testing of the system has been completed, authorization for operation will be requested from the ISIN. [86]

4.2.4 Trino

The Trino nuclear plant was a 270 MW_{el} PWR shut down in 1987 on the aftermath of the referendum. In 1999 SOGIN acquired it in order to decommission it with the objective to reach the brown field in 2029. [87]
All the irradiated nuclear fuel resulting from the operation of the plant (487 fuel elements) was sent abroad for reprocessing in several stages. The residues originating from the reprocessing operations, conditioned in the glass matrix, will return to Italy to be temporarily transferred to the Deposito Nazionale. [69]
During the operation of the plant, the wet cooling towers allowed to cool the water of the secondary circuit at the outlet. The dismantling of the electromechanical components was carried out, with the removal of approximately 160 tons of ferrous material, 61 of plastic and 40 of cables. [88]
Between 2003 and 2005 the turbines were dismantled and the materials produced were characterized, removed from the site as conventional materials and then destined for recovery. Some systems are still in operation in the building, which are necessary in the decommissioning phase. In 2018, the project for the partial dismantling of the building was started. [89]
In the plant there are two temporary deposits for radioactive waste and a temporary buffer, the test tank, which house all the previous waste present on the site inside containers of different types. Restructuring of the two warehouses is planned, in order to optimize the storage spaces, allowing them to house both the waste currently present and those that will be produced by decommissioning. In addition to the civil and plant renovation of the two buildings, campaigns have been underway for the treatment and repackaging of radioactive waste since 2012, to make them suitable for transfer to the Deposito Nazionale.

In order to proceed with the adaptation of the first of the two deposits, 300 380-liter overpacks were temporarily transferred to the local buffer test tank during 2018. By 2019, the tender process will be launched for the adaptation works of this temporary storage, whose completion is scheduled for 2022. Once the refurbishment work has been completed, it will be possible to fill it with the radioactive waste currently stored in the test tank and in the other temporary deposit. [90]
The treatment of radioactive resins will take place through Wet Oxidation Technology (WOT), in a special plant. This technique, used in the conventional field for organic waste, has the advantage of producing a reduced environmental impact and can be easily reused for the treatment of similar residues in other plants. The objective of oxidation is to reduce the volume and organic load of the resins through a three-step process.

1. Pre-treatment: recovering of the resins exhausted by the purifiers, dividing them into homogeneous groups.

2. Treatment: through wet oxidation, the organic matter is transformed into water and carbon dioxide, and the inorganic matter into a residue composed of insoluble oxides and soluble salts.

3. Post-treatment: concentration of the residue treated before it is sent to the conditioning system. [91]

Between 2003 and 2004 the decontamination of the steam generators of the circuits was carried out, with a 100-fold reduction in contamination levels. Between 2005 and 2008, asbestos was removed from the pipes, the pressurizer and the steam generators of the reactor building. The ventilation system of the building was adapted between 2009 and 2013, creating a dynamic air exchange system suitable for the decommissioning phase. In the same years, the access routes to the container were restructured, to allow the handling of materials and the passage of personnel in the dismantling phase. Between 2013 and 2017 all the non-contaminated components of the reactor were removed, producing about 200 tons of iron potentially destined for recovery after systematic checks of the radiological status. Between 2016 and 2017, samplings were carried out with the purpose of evaluating the radiological status of the plant. This led to the collection of about 60 metal samples currently undergoing radiochemical analysis. In 2015, the actual dismantling activities began, with the removal of the fuel loading crane, the anti-missile screen, the asbestos placed on the head of the vessel and other accessories that interfered with the opening of the vessel. The activities continue with the re-commissioning of all accessory systems necessary for opening the vessel (overhead cranes, tank circulation systems, water accumulation systems). These activities include the replacement of the reactor cavity level control systems, the verification of the integrity of the welds and the replacement of some obsolete components. Preparation for the tender for the dismantling of the vessel is underway. Firstly, the radiological characterization of the vessel and the internals will be performed. At the end of this activity, it will be possible to proceed with the dismantling of the vessel head and the upper part of the reactor, called upper package. The removal and dismantling of the activated internals and the cylindrical body will then take place. Parallel to the activities on the vessel, the dismantling of the primary circuit will be carried out. In the first phase, the primary will be isolated from the vessel. Subsequently, the primary pumps, primary circuits and auxiliary systems will be dismantled. In the third phase, the large components

(pressurizer and steam generators) will be dismantled. [92]
In the premises that housed the ventilation systems during operation, in 2016 the removal of the accumulators and the components of the emergency cooling system took place. The materials management station will be built in the same building, with the aim of minimizing radioactive waste, through the decontamination of the maximum amount of materials present in the plant to allow their release as conventional material. Inside it will be carried out operations of treatment, characterization, decontamination, cutting and reduction of the volumes of the materials. [93]

4.2.5 Bosco Marengo

The Bosco Marengo plant produced the fuel elements for Italian and foreign nuclear plants. Starting in 1987, with the closure of the Italian nuclear program, the plant has gradually diversified its activities and in 1995 nuclear activities were stopped. In 2005 the SOGIN became the owner of the plant with the aim of carrying out its decommissioning. [94]
In the nuclear fuel production plant the manufacturing activities for the pads that make up the fuel elements took place. In 2009 the SOGIN started the decommissioning activities of the plant. In particular, the dismantling of the fuel production plant ended in 2011. Subsequently, from 2012 to 2018, the auxiliary services of this plant were dismantled (ventilation system, decontamination tank and drainage systems for liquid radioactive effluents). The activities were carried out using both wet and dry decontamination techniques of the materials to be treated and produced approximately 376 tons of metal material that was possible to release from the site without any radiological constraints. [95]
The adaptation of the B106 room as a temporary storage facility, allows for the storage of all radioactive waste from the site, pending their transfer to the Deposito Nazionale. In 2017, the adaptation works of the B106 room began, while the radioactive waste is stored in an adjacent building which, in 2011, was upgraded to a temporary buffer station and entered into operation in 2012, first hosting only non-combustible radioactive waste and then, from April 2015, all other waste previously stored in other areas of the site. The adaptation works of the B106 to temporary storage ended in the first half of 2019. Functional testing of all plants and systems in the B106 warehouse is currently underway. In this regard, in February 2021 the testing activities on the electrical system, drainage system, Heating, Ventilation and Air Conditioning (HVAC) systems and fire detection were successfully completed in the presence of the ISIN. Once the tests have been completed, the investigation will begin to obtain the Operating License of the B106 as a temporary deposit. The entry into operation of the B106 will allow to arrange all the radioactive waste of the site, pending their transfer to the future Deposito Nazionale. [96]
The plant solid radioactive waste treatment ended in the first quarter of 2019. During the year, in fact, 390 220-liter drums containing solid radioactive waste were produced, in addition to 902 drums produced in the decade 2009-2018 by the dismantling activities. Almost all of the drums produced so far have been treated, supercompacted and conditioned by cementation. Instead, blank tests were carried out for the treatment of the small amount (about 2 m^3) of very low (VLLW) and low activity (LLW) liquid radioactive waste produced during the operation of the plant. At the end of the tests and after obtaining the approval of the operational plan from the Control Authority, the waste will be made

suitable for transport through a solidification process using a special polymer and cement powder. The final artifacts will return for temporary storage in Bosco Marengo pending their transfer to the Deposito Nazionale. [97]

4.2.6 Casaccia

Within the ENEA Research Center in Casaccia, the SOGIN has been managing the OPEC plant (hot operations) and the IPU plant (plutonium plant) since 2003 with the aim of reaching the brown field in 2029. OPEC-1 was the first plant in Italy to perform post-irradiation research and analysis on nuclear fuel elements. Today it is a temporary deposit where the historical inventory of irradiated materials transferred here from the various research chains for destructive tests in the cell is kept. Adjacent to OPEC-1, OPEC-2 was built to expand the nuclear research, control and analysis activities that were carried out in OPEC-1, but it never entered into operation. Today OPEC-2 has been refurbished for the temporary storage of radioactive waste. The IPU plant carried out research activities on the technologies for the production of nuclear fuel elements. In 1990, with the closure of the Italian nuclear program, research activities were stopped. [98]
The Glove Boxes of the IPU plant are confined environments that, during the exercise, were used to manipulate plutonium as part of research activities for the production of nuclear fuel elements. The 56 Glove Boxes are divided into four levels of complexity, linked to size and content, as well as the design and operational difficulties of their dismantling. This project, which represents the most significant activity in the field of plant decommissioning, provides for a first phase of planning the interventions and the acquisition of equipment such as, for example, cutting equipment, accessories for handling the Glove Boxes and the containment curtains with α seal. The latter are equipped with gloved passages that allow SOGIN operators to carry out dismantling operations from the outside. The operations include the preliminary remediation of the Glove Boxes, the preparation of the dismantling station, the handling and introduction of the Glove Boxes into the containment tent, the dismantling of the Glove Boxes and of the used tent and the management of the radioactive waste produced. The first Glove Boxe was dismantled in 2010. From 2012 to 2014, after obtaining the necessary authorizations from the Control Authority, all first and second level Glove Boxes were dismantled. In 2016, the third level ones were dismantled and the dismantling operations of the fourth level ones started, with greater complexity in terms of size and content, which will end, according to the current program, in 2021. At the end of 2020, 53 Glove Boxes have been dismantled. [69] [99]
The OPEC-2 deposit, with a storage capacity of over 2300 285-liter stainless steel drums, was built for the temporary storage of solid radioactive waste deriving from the operation of the IPU and from subsequent dismantling activities, including those produced by the Glove Boxes. The plant works, which began in 2013, ended in 2017 with the final testing of the security systems of the deposit. After obtaining the operating license in 2018, in September 2019 the authorization for the first loading campaign of the OPEC-2 temporary deposit was issued by the Security Authority, which allowed the start of the disposal of the IPU plant previous radioactive waste. [100]
During the research activities carried out at the IPU plant, less than 1 m^3 of liquid radioactive waste contaminated with plutonium was produced. These wastes, both of organic and aqueous matrix, are stored in part in the IPU plant and in part in Nucleco's temporary

deposits. The definition of the treatment and conditioning processes of these types of waste requires very in-depth studies and tests. The feasibility analyses to treat and condition the organic liquid waste are still in progress, while for the aqueous radioactive waste the qualification activities of the conditioning process through cementation in an innovative Glove Box, that will be installed in the IPU plant, were completed in December 2018. [101]
A plant called Waste Management Facility (WMF), designed for the treatment of solid radioactive waste contaminated by plutonium, will be built. In particular, both radioactive waste already present at IPU and those that will come from future plant dismantling activities will be treated in the WMF. [102]

4.2.7 Rotondella

The Impianto di Trattamento e Rifabbricazione Elementi di Combustibile (ITREC) is located within the ENEA Trisaia Research Center in Rotondella. Between 1968 and 1970, 84 irradiated uranium-thorium fuel elements from the Elk River experimental reactor (Minnesota) were transferred to the plant. In 1987, following the referendum on nuclear power, activities were stopped. Since then, safe maintenance has been guaranteed. In 2003 the SOGIN took over the management of the plant with the aim of carrying out its decommissioning. [103]
In 2005 a chemistry and radiochemistry laboratory was created with the aim of autonomously managing the analysis of the samples required by the various environmental radiological monitoring plans that the site is required to comply with. The laboratory measures α, β and γ emitting radionuclides in the various environmental matrices such as atmospheric particulate, groundwater, sea water, milk, fruit, vegetables, soil, etc. for a total of about 2000 determinations per year and is also equipped with the necessary equipment for monitoring the internal contamination of exposed workers pursuant to Legislative Decree n° 230/1995. [104]
The Impianto Cementazione Prodotto Finito (ICPF) is the plant for cementing the liquid uranium-thorium solution (about 3 m^3) called finished product and resulting from the experimental activities of reprocessing of the fuel that took place during the operation of the plant. The radioactive liquid solution derived from the process was stored in a stainless steel tank, the W120, positioned inside a concrete cell in the Waste 1 area. The ICPF project involves the construction of a process building, which will house the remote systems for cementing the liquid solution, the finished product, and a temporary deposit. Inside the latter, the artifacts containing the cemented waste will be safely placed and a dedicated area will house the two casks with the 64 fuel elements currently stored in the plant pool. In 2004 the preparatory activities for the construction of the plant began, such as the design and qualification of the cement matrix, in which to incorporate the finished product. Between 2007 and 2010 the prototype (mock-up) of the cementation cell was created, in scale 1:1, necessary to qualify the product, to test the components and the cementation process and to train the personnel to be involved in the activities. Between 2014 and 2017 the construction site was started with the excavation works and the construction of a piling to support the surrounding land, the works were carried out foundation of the temporary deposit and some partial elevation works. It is estimated that the construction work for the temporary deposit will be completed in 2023 and those for the trial building

in 2024. [105]
At the end of the 1960s, in the ITREC in Rotondella a special underground structure, called Fossa 7.1, was designed and built to dispose of ILW deriving from the reprocessing activities of the fuel elements. It is a vertical structure with a prismatic shape with a mass of about 130 tons and a volume of 54 m^3, today called a monolith, which is located at a depth of 6.5 meters. Inside, the radioactive waste is stored in 220 liter oil-type drums, embedded in cement mortar, inside four square section wells. In 2007, the hydraulic containment barrier was built to ensure maximum safety conditions in carrying out the reclamation works of Fossa 7.1. The first remediation operations, started in 2012, involved civil works, the construction of a shed to ensure adequate confinement of the construction site from the external environment and the installation and testing of service systems. In 2013, the roof structure of the area where Fossa 7.1 is located was completed to ensure maximum safety in every phase of the reclamation work. Besides, the ventilation, fire and electrical systems, and radiological monitoring systems were built. In 2014, once the stabilization, drainage and protection system of the piling was implemented, excavation works and structural investigations on the monolith began, during which a small percolation of aqueous liquid from one wall of the monolith occurred, which has concerned a limited area of land equal to about 2 m^2. The SOGIN has implemented all the necessary safety measures and has ensured the sealing of the percolation area, with the immediate start of the work to remove the affected soil and to collect samples of the leachate liquid and the soil itself. All the radioactivity values found and the consequent radiation protection assessments excluded consequences for the environment, the population and workers. The works therefore continued with the drainage of about 800 liters of liquid found inside the monolith. Before starting the cutting operations, further stabilization systems of the monolith were implemented. In March 2019, work began on cutting the monolith in a confined environment. Firstly, a horizontal cut was performed, perforating the base of the structure using a core barrel with disposable tips. These transactions concluded in May 2019. The vertical cutting was then carried out with diamond wire from top to bottom, separating the four wells from each other. The activity ended in September 2019. Subsequently, the individual wells were encapsulated to proceed with their lifting and extraction, to then be transferred in maximum safety to a temporary deposit on the site. During an inspection carried out following the positioning of the third well extracted, drops of liquid were detected on the floor, an eventuality expected and evaluated in the project of the activities. The SOGIN promptly notified the event to the ISIN Control Authority which, during a specific inspection, took note of the limited nature of the extent of the spill and the absence of radiological consequences for the population and the environment. During the inspection, environmental samples were acquired for independent radiometric determinations that were carried out by the ARPA Basilicata, which confirmed the absence of radiological consequences for the population and the environment as the values detected are lower than the reference levels indicated by the Supervisory Body. The removal of Fossa 7.1 monolith was completed on 18^{th} December 2019, with the extraction of the last well. In order to finish the remediation activities of the area that housed the monolith, in 2020 the characterization for the management of materials according to the implant procedures and sampling required by the Characterization Plan to assess the radiological status of the area were carried out. [69] [106]

A project called SIstemazione RIfiuti Solidi (SIRIS) concerns the treatment of solid radioactive waste present in ITREC. Between 2005 and 2011, 852 drums containing solid radioactive waste produced by past plant safety maintenance activities were characterized, treated and conditioned, which were placed inside 21 containers. The activities took place inside the specially set up cutting cells and, at the end, the containers were removed and the areas freed. As part of the SIRIS project, work continues on the characterization, treatment and conditioning of solid radioactive waste that derive from the activities of maintaining the plant in safety and from the preparatory activities for the decommissioning of the site. [107]

4.2.8 Saluggia

In the Enriched URanium EXtraction (EUREX) plant in Saluggia research activities were carried out on the reprocessing of irradiated nuclear fuel. The activities were interrupted in 1984 and the decommissioning started in 2003 when the SOGIN acquired it. The brown field should be reached in 2035. [108]
In this plant there are a total of about 270 m^3 of radioactive liquid waste, of which 125 m^3 with the highest level of activity. The latter between 2008 and 2009 were transferred to the New Tank Park, a structure built between 2004 and 2006 to ensure maximum safety in the storage of this type of waste. [109]
The pool of the EUREX plant housed the fuel elements that were the subject of the reprocessing campaigns. In 2004, following the discovery of the partial loss of containment of the pool, the SOGIN decided to start the emptying and reclamation operations. The results of the environmental investigations, carried out by the relevant bodies, showed that the loss did not cause any impact on the population and the environment. The works were completed in 2008 and were divided into several phases: removal of the components present in the pool, removal of the irradiated fuel inside special containers, cleaning of the bottom, emptying and decontamination of the water. [110]
In 2011, work began on the construction of the temporary storage D2, necessary to house LLW and ILW. It will store approximately 2,500 m^3 of conditioned radioactive waste from previous and future activities. The warehouse, whose works ended in 2015, was authorized for operation in March 2019. [111]
The Saluggia site contains a total of approximately 300 m^3 of radioactive liquid waste, which mainly comes from the reprocessing campaigns of irradiated fuel elements conducted in the 1970s and 1980s. To solidify the liquid waste, the CEMentazione EurEX (CEMEX) facility will be built, within which the waste will be cemented and conditioned. CEMEX will be built in an area adjacent to the New Tank Park, from which the pipes for the transfer of radioactive liquids to the cementation plant will start. The storage of conditioned artifacts will take place in the adjoining temporary warehouse, D3. In 2019 the construction site was restarted with the provisional installation of the scaffolding, the cleaning of the reinforcing bars already present in the building and the qualification of the special cement matrix that will be used for the construction of the structural works. The activities concerned the completion of the load-bearing structures of which the deposit is made up: the handling area of the artifacts, maneuvering room, health physics rooms, construction and waterproofing of the floor. The structural civil works were completed in March 2020.

Subsequently, on 14^{th} July 2020, a new call for tenders for the completion works of the CEMEX facility was published with a notice in the Official Journal of the EU. The activities foreseen by the tender concern the construction of the process building, where the liquid radioactive waste will be cemented, and the installation in the annexed temporary storage D3 of the equipment and auxiliary systems for control and handling of medium-sized artifacts. The total value of the activities envisaged based on the tender is € 128.5 million, to be carried out by 2023. Subsequently, the tests of the plant will be carried out aimed at its entry into operation. [112]
At the EUREX plant in Saluggia, about 15 m^3 of low-radioactive organic liquid radioactive waste from previous site activities are stored in an underground tank. The program for their treatment was started in 2017 with the creation of a system to take the samples necessary for their characterization and for the development of the project of the plant for the extraction and immobilization of waste in special drums. This waste, which due to their low radioactivity and organic nature will not be conditioned by solidification in the CEMEX complex, will undergo a heat treatment at a qualified foreign operator, with subsequent conditioning of the residual ashes. For the purposes of transport to the foreign plant, the waste must be previously immobilized in solid form inside suitable containers. Immobilization must be carried out with material that can be destroyed by heat treatment without causing a significant volumetric increase in the residue. The extraction and immobilization operations will be carried out within a light confinement structure, with both static and dynamic insulation characteristics, made with self-supporting modular panels of reinforced fiberglass. At the end of the works the structure will be dismantled. The extraction of organic liquid waste will take place through a suction system that will allow the passage from the tank to the drums where the liquid will be stabilized through special absorbent polymers. The drums will then be stored in containers within a specific area of the site before being sent to the foreign operator. It is estimated the production of 20 final artifacts with the ashes incorporated in cement mortar, for a total gross volume of 11 m^3. The final artifacts will return to the EUREX plant where they will be stored in view of their final delivery to the Deposito Nazionale. The activities of liquid extraction, immobilization and incineration heat treatment will be started only after obtaining all the necessary authorizations and will last approximately 24 months. [113]
A WMF will be built for the treatment and conditioning of radioactive waste produced by previous activities and the entire decommissioning program. This plant will process the solid waste of Saluggia creating a finished product that complies with the acceptability requirements of the Deposito Nazionale and therefore with an adequate safety status for temporary storage in the site's depots. The plant provides two distinct processes of cutting/volume reduction and characterization, one for ILW, and the other for LLW and VLLW. In addition, large components and different materials, especially concrete and steel, will be processed. [114]

4.2.9 Ispra

In 2018, the Italian Government with the 2018 Budget Law entrusted the SOGIN with the decommissioning of the Ispra-1 reactor located in the complex of the Joint Research Center (JRC) of the European Commission in Ispra, aiming to reach the brown field in 2034. The decommissioning operations of the Ispra-1 reactor are planned in three phases:

preliminary activities, decommissioning of the reactor and final reclamation of the site. These activities will be launched only after the approval by the ISIN Control Authority of the first phase dismantling application, presented by the SOGIN on 29^{th} April 2020.
In 2020 the SOGIN started the project that will lead to the emptying of the pool containing about 200 m^3 of water as its first activity. This operation follows the removal of activated metal components and metallic and muddy sediments, completed in recent years by the JRC of Ispra. The emptying of the pool involves filtration and radiological purification of the water present through a special filtration and treatment system, based on the selectivity of the ion exchange resins for the radionuclides still present, already successfully adopted by the SOGIN for the reclamation of the pool of the EUREX of Saluggia. The emptying operations, started in February 2021, are carried out progressively in batches of about 5 m^3 each, a volume proportionate to the receptive capacities of the site's liquid effluent treatment plant. The final water discharge will take place in compliance with the site's discharge formula. The conclusion of the activity is expected in 2024. [69] [115]

4.3 The Deposito Nazionale and the Parco Tecnologico

The Deposito Nazionale will be a surface environmental infrastructure that will make it possible to definitively arrange the radioactive waste, now stored in dozens of temporary deposits in the country, produced by the operation and dismantling of nuclear plants and by the daily activities of nuclear medicine, industry and research. The Deposito Nazionale will consist of structures for the disposal of VLLW and LLW, and those for the temporary storage of ILW and HLW, which must subsequently be transferred to a geological repository suitable for their final accommodation. Together with the Deposito Nazionale, the Parco Tecnologico will be created, a center for applied research and training in the field of nuclear decommissioning, radioactive waste management and radiation protection, as well as environmental protection.
The Deposito Nazionale and the Parco Tecnologico will occupy a surface of about 150 hectares, of which 40 hectares are dedicated to the Parco Tecnologico and 110 hectares to the Deposito Nazionale, as shown in Figure 4.1. [116]
It will allow the definitive settlement of approximately 78,000 m^3 of LLW and the temporary storage of approximately 17,000 m^3 of ILW and HLW. About 60% of the 95,000 m^3 that will be delivered to the Deposito Nazionale will derive from the operation and decommissioning of nuclear plants, while the remaining 40% will come from nuclear medicine, industrial and research activities. [118]

4.3.1 Italian radioactive waste

The Italian radioactive waste classification refers to the Ministerial Decree of 7^{th} August 2015 "Classification of radioactive waste, pursuant to Art. 5 of the legislative decree 4^{th} March 2014 n° 45".
The VSLW includes radionuclides with half life lower than 100 days and are generated by medical application and research. They will end up in a temporary storage (art. 33

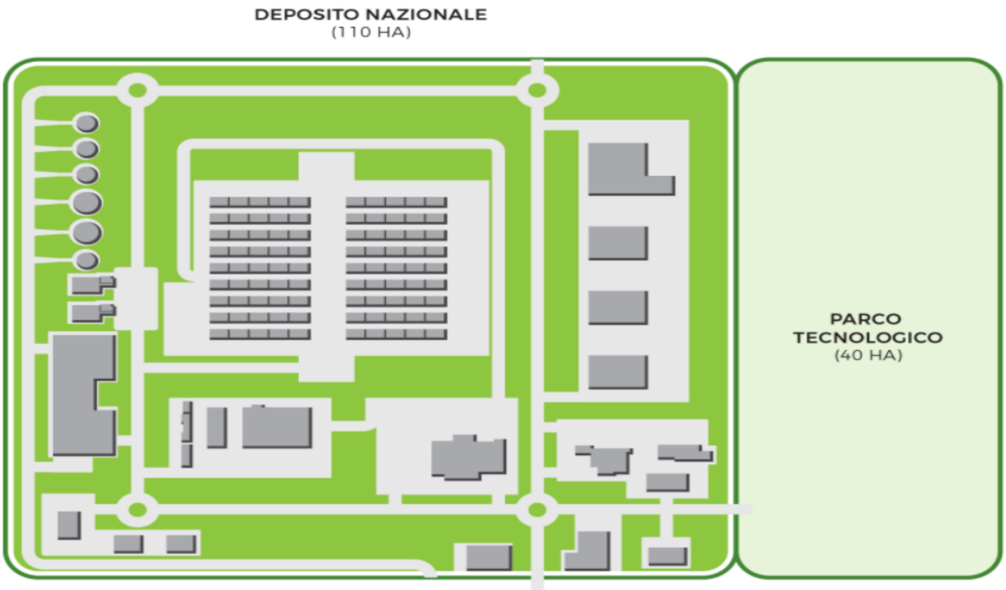

Figure 4.1. Surfaces occupied by the Deposito Nazionale and the Parco Tecnologico. [117]

Legislative Decree n° 230/1995 and disposal in compliance with the provisions of the Legislative Decree n° 152/2006. [119]
The VLLW consider materials coming from safety activity of maintenance and from decommissioning activities of nuclear installations and it has radioactivity lower than 100 $\frac{Bq}{g}$ (with α-emitters contribute lower than 10 $\frac{Bq}{g}$). If this condition is reached in less than ten years its final disposal is the same as for VSLW, otherwise it will be sent to the Deposito Nazionale (when ready).
LLW includes short lived radionuclides with radioactivity lower than 5 $\frac{MBq}{g}$ and long lived radionuclides with radioactivity lower than 400 $\frac{Bq}{g}$. It comes from nuclear installations and it will go to the Deposito Nazionale.
ILW includes short lived radionuclides with radioactivity greater than 5 $\frac{MBq}{g}$ and long lived radionuclides with radioactivity greater than 400 $\frac{Bq}{g}$. It does not generate heat and it comes from decommissioning of plants related to nuclear fuel cycle and from research laboratories. If it has α-emitting radionuclides with radioactivity lower than 400 $\frac{Bq}{g}$ and β-γ emitters with concentrations which meet the radiation protection objectives established for the surface disposal facility, it will be sent to the Deposito Nazionale. On the other hand, if its radionuclides have concentrations which do not comply with the radiation protection objectives established by the surface disposal facility, it will go to the temporary storage facility of the Deposito Nazionale waiting for the geological disposal.
HLW is waste which generates heat or has high concentrations of long-lived radionuclides or both. It will be sent to the temporary storage facility of the Deposito Nazionale waiting for the geological disposal. [120]
Italy's waste inventory is reported in Table 4.1. It is updated to 31^{st} December 2020. [69]

Type of waste	VLLW	LLW	ILW
	m^3	m^3	m^3
Latina			
Conditioned	18	2	89
Unconditioned	768	639	333
Total	786	641	422
Caorso			
Conditioned	103	8	0
Unconditioned	725	980	0
Total	828	988	0
Garigliano			
Conditioned	55	921	90
Unconditioned	1,618	221	0
Total	1,673	1,142	90
Trino			
Conditioned	35	78	3
Unconditioned	954	143	62
Total	989	221	65
Bosco Marengo			
Conditioned	164	323	0
Unconditioned	8	5	0
Total	172	328	0
Casaccia			
Conditioned	0	0	0
Unconditioned	0	3	460
Total	0	3	460
Rotondella			
Conditioned	882	220	163
Unconditioned	1,775	454	31
Total	2,657	674	194
Saluggia			
Conditioned	298	86	34
Unconditioned	1,128	547	531
Total	1,426	633	565
Ispra			
Conditioned	0	0	0
Unconditioned	90	3	1
Total	90	3	1

Table 4.1. Italian radioactive waste inventory. [69]

4.3.2 Technical characteristics of the Deposito Nazionale

The Deposito Nazionale will consist of a structure with engineering barriers and natural barriers placed in series for the containment of radioactivity, designed on the basis of the best international experiences and according to the IAEA and the ISIN standards. The protective engineering barriers will be built with specific reinforced concrete conglomerates, guaranteed to confine the radioactivity of the waste for the time necessary for its decay to levels comparable to the ranges of variation of environmental radioactivity.
The artifacts (the first barrier) are the structures, cylindrical or parallelepiped, made up of metal containers and the radioactive waste inside them, already conditioned in a solid form. The chemical and physical stability allows the product to be handled and transported safely. The artifacts will then be placed inside special concrete modules.
The modules are the second barrier and they are parallelepiped-shaped structures (3 m x 2 m x 1.7 m) in special concrete, reinforced or fiber-reinforced, ensure their resistance for over 350 years. After having placed the artifacts, they are cemented together with a special mortar. A lid, also made of special concrete, will seal the module before it is placed in the cell.
The cells represents the third barrier and they are the buildings in special reinforced concrete (27 m x 15.5 m x 10 m), designed to resist for at least 350 years, where the radioactive waste will be permanently placed. Within the Deposito Nazionale, 90 cells will be built, organized in juxtaposed rows, which outline the actual area in which to permanently arrange the radioactive waste. The engineering barriers of the Deposito Nazionale and the characteristics of the site where it will be built will ensure the isolation of radioactive waste from the environment for over 300 years, until its decay to levels that are negligible for human health and the environment. About 78,000 m^3 of VLLW and LLW will be permanently placed in the cells. Once all the cells have been filled, they will be covered with a multilayer hill.
The multilayer hill, the fourth barrier within the Deposito Nazionale, is an artificial structure arranged to cover the cells. It is made with layers of different materials, for a total thickness of a few meters, in order to prevent the entry of water into the deposit, drain rainwater, isolate waste from the environment and improve the visual impact of the structure. The Deposito Nazionale, once its reception capacity has ended, will be closed and will last at least 300 years, and will then be released without any radiological constraints. Figure 4.2 shows the conceptual design of the four barriers.
A complex of buildings suitable for the long-term storage of the 17,000 m^3 of ILW and HLW will be built in a specific area of the Deposito Nazionale, which will remain temporarily in the deposit, to then be permanently placed in a geological deposit. [116]

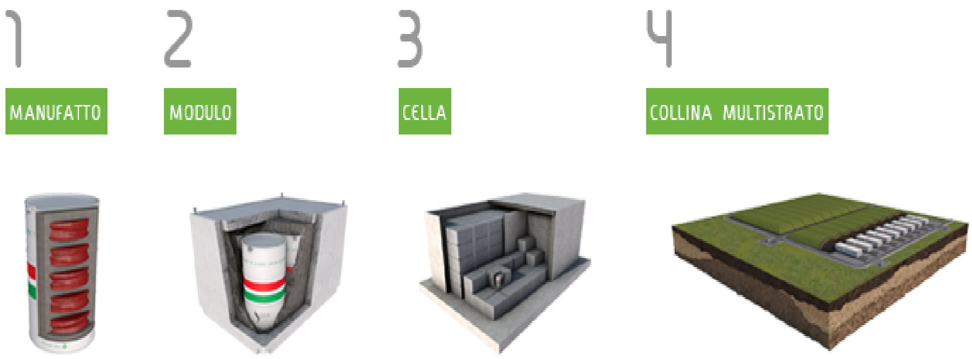

Figure 4.2. The Deposito Nazionale barriers. [121]

4.3.3 Life stages

There are three life stages of the Deposito Nazionale: operation, closure and institutional control.
During the operation phase the waste is received from the Deposito Nazionale as artifact and inserted into a module and then permanently placed inside a cell cells. At the end of its filling, each cell will be sealed and waterproofed. The operating phase will last approximately 40 years. In the first few years, the waste deriving from the dismantling of nuclear installations will be delivered to the Deposito Nazionale and subsequently that produced by medical, industrial and research activities.
After filling the cells, the closing phase begins, during whose activities will be the construction of the multilayer roof, the progressive removal of ILW and HLW, the dismantling of the plants used for the disposal of VLLW and LLW, and of the buildings for temporary storage of ILW and HLW. The environmental and radiological monitoring network will also be completed in this phase.
Once the filling is complete, the Deposito Nazionale will be closed and will enter the institutional control phase, during which a drainage system, installed under each cell, will ensure the collection and treatment of the water deriving from any infiltrations or condensation inside the cells. The facility will also be monitored to prevent intrusions and ensure maximum efficiency of the barriers and the environmental and radiological monitoring network will remain operational. The institutional control phase will continue for about 300 years, after which, thanks to the decay of radioactivity and on the basis of a long-term safety analysis (Safety Assessment), the site will be freed from radiological constraints, making it available for other uses. [122]

4.3.4 Localization process

The Legislative Decree n° 31/2010 punctually describes the process of locating the Deposito Nazionale and the Parco Tecnologico. In Figure 4.3 all the stages of this process are shown.
The criteria developed by the Istituto Superiore per la Protezione e la Ricerca Ambientale

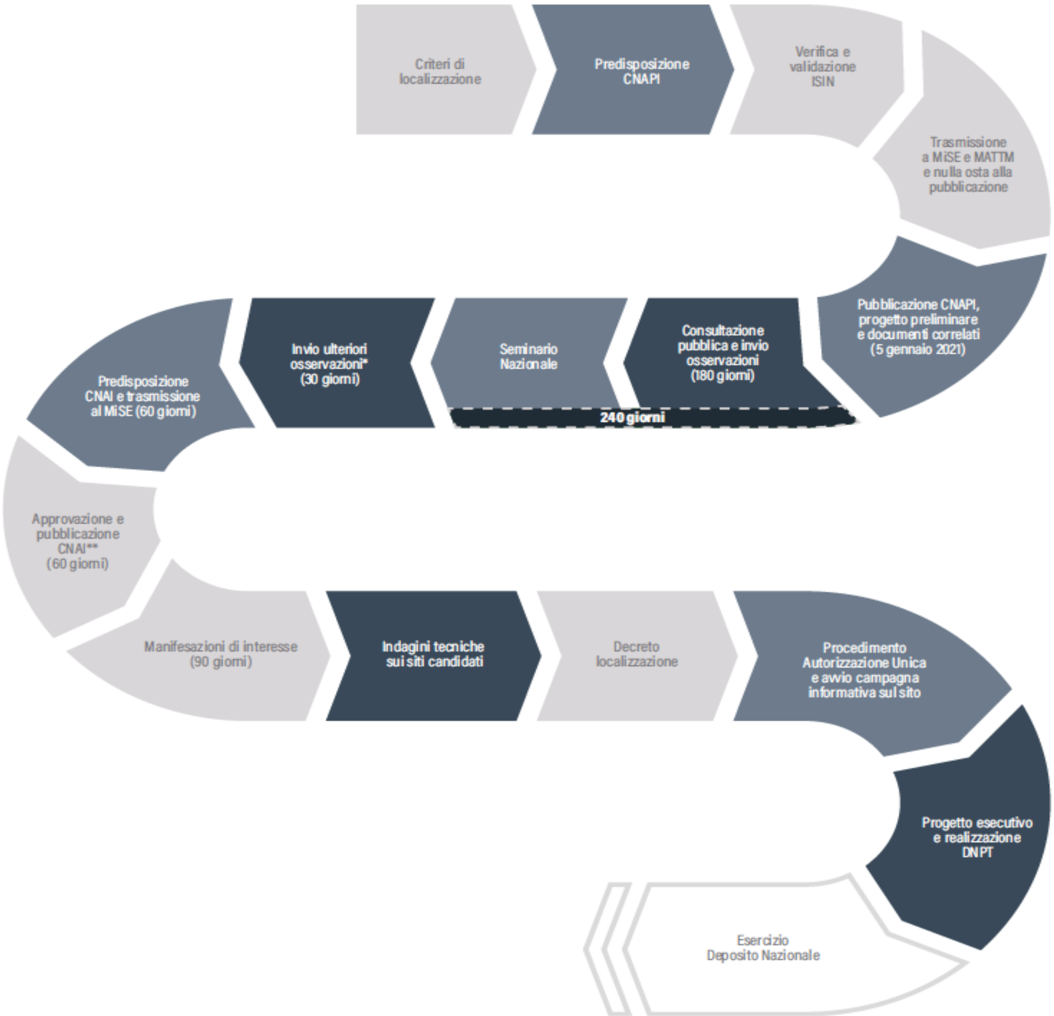

Figure 4.3. The various steps of the localization process. [69]

(ISPRA), now ISIN, in the Technical Guide n° 29, in line with the standards of the IAEA, represent a set of fundamental requirements and evaluation elements to arrive, with a progressive level of detail, in identifying the areas potentially suitable for hosting the Deposito Nazionale. The criteria were formulated to identify areas where the integrity and safety of the National Deposit is guaranteed over time. In particular, 15 Exclusion Criteria are set to exclude areas of the national territory whose characteristics do not allow to guarantee full compliance with safety requirements, and 13 Investigation Criteria, to evaluate the areas identified following the application of the Exclusion Criteria. The application of the Exclusion Criteria leads to the identification of potentially suitable areas. It is carried out through checks based on regulations, data and technical knowledge available for the entire national territory, also through the use of GIS-Geographic Information Systems and, in some cases, databases managed by entities public. [123] These criteria exclude:

1. Active or dormant volcanic areas.
2. High seismicity areas.
3. Areas affected by faulting phenomena.
4. Areas characterized by geomorphological and/or hydraulic risk and/or danger of any degree and river belts.
5. Areas characterized by the presence of alluvial deposits of the Holocene age.
6. Areas located at an altitude greater than 700 m a.s.l.
7. Areas characterized by slopes with an average gradient greater than 10%.
8. Areas up to 5 km away from the coastline or located in a greater distance but at an altitude of less than 20 m a.s.l.
9. Areas affected by the karst morphogenetic process or with the presence of sudden catastrophic sinking.
10. Areas characterized by rising piezometric levels or which, in any case, may interfere with the foundation structures of the Deposito Nazionale.
11. Protected natural areas identified in accordance with current legislation.
12. Areas that are not at an adequate distance from populated areas.
13. Areas that are less than 1 km away from motorways and main extra urban roads, and fundamental and complementary railway lines.
14. Areas characterized by the known presence of important underground resources.
15. Areas characterized by the presence of industrial activities at a relevant risk of accidents, artificial dam and hydraulic barriers, airports or operational military firing ranges. [124]

The application of the Investigation Criteria can lead to the exclusion of further portions of the territory within the potentially suitable areas and to the identification of sites of interest. It is carried out through specific investigations and assessments on the areas that have not been excluded. [123] The aspects to be carried out are:

1. The presence of secondary volcanic manifestations.
2. The presence of significant vertical movements of the soil as a consequence of phenomena of subsidence and uplift (tectonic and/or isostatic).
3. The geological-morphostructural structure and the presence of lithotypes with vertical and lateral heteropy.
4. The presence of endorheic water basins.

5. The presence of accelerated erosion phenomena.
6. The weather-climatic conditions.
7. The physical-mechanical parameters of soils.
8. The hydrogeological parameters.
9. The chemical parameters of the soil and groundwater.
10. The presence of habitats and animal and plant species of conservationist importance, as well as geosites.
11. Agricultural productions of particular quality and typicality and places of archaeological and historical interest.
12. The availability of primary communication routes and transport infrastructures.
13. The presence of critical, significant or strategic infrastructures. [124]

On 4^{th} June 2014 these siting criteria were sent by the ISPRA (now ISIN) to the SOGIN which used them to prepare the Carta Nazionale delle Aree Potenzialmente Idonee (CNAPI) which was sent to the ISPRA (now ISIN), to the Ministry for Economic Development and to the then Ministry of the Environment and the Protection of Land and Sea (now Ministry of the Ecological Transition) on 2^{nd} January 2015. According to the Technical Guide n° 29, the CNAPI means areas, even large ones, which present characteristics favorable to the identification of sites capable of being suitable for the location of the Deposito Nazionale, through subsequent detailed investigations and on the basis of the results of safety analyses conducted taking into account the design features of the Deposito Nazionale structure. [125] After the verification made by the ISIN, by the Ministry for Economic Development and by the then Ministry of the Environment and the Protection of Land and Sea (now Ministry of the Ecological Transition), the CNAPI was published on 5^{th} January 2021. Overall, in the CNAPI proposal, 67 potentially suitable areas have been identified today, of which only one will be chosen, at the end of the localization process, as the only national site suitable for hosting the Deposito Nazionale. The 67 areas are grouped into four sets with decreasing suitability order (A1, A2, B and C). This order characterizes every potentially suitable area from the point of view of logistical and infrastructural efficiency. [126]
The CNAPI proposal is subject to public consultation. In the 180 days following the publication, the Regions, local authorities and qualified stakeholders were able to formulate and transmit comments and technical proposals to the SOGIN in written and non-anonymous form. This first phase of public consultation ended on 5^{th} July 2021. Within 240 days from the start of the public consultation, the SOGIN promotes the National Seminar in which qualified stakeholders are invited to participate in order to deepen all the technical aspects relating to the Deposito Nazionale and the Parco Tecnologico and the compliance of the areas identified with the requirements of the Technical Guide n° 29. The seminar also examines the aspects related to the safety of workers, the population and the environment and the possible economic and territorial development benefits associated with the construction of the work.

After the National Seminar, which will end on 15^{th} December 2021 with the publication of the overall report of the works which will end on 24^{th} November 2021, the legislation will provide for a second phase for the formal transmission of further observations and the drafting by the SOGIN of the Carta Nazionale delle Aree Idonee (CNAI) and it will send it to the Ministry of the Ecological Transition. [127]
Having acquired the technical opinion of the ISIN, the Ministry of Ecological Transition definitively approves it, in agreement with the Ministry of Infrastructure and Transport (now Ministry of Sustainable Infrastructure and Mobility). The definitive version of the CNAI is therefore the result of the integration into the CNAPI of the contributions that emerged and agreed upon in the various phases of the Public Consultation. Based on these opinions, the Ministry of Economic Development validates the final version of the CNAI. [128]
After that, the SOGIN will open the next phase of confrontation aimed at collecting voluntary and non-binding expressions of interest from the Regions and local authorities whose territory also partially falls within the areas suitable for hosting the Deposito Nazionale and the Parco Tecnologico. In the event that no expressions of interest are expressed by local authorities, or if all those received are withdrawn at a later time, the SOGIN will have to promote bilateral negotiations with the Regions in whose territory the suitable areas fall. In case of failure of the bilateral negotiations (lack of agreement), an interinstitutional table will be convened, as a further attempt to reach a shared solution. The goal of a public consultation thus conceived is to develop, in different stages and with different tools, a shared path for reaching an agreement. Having reached an agreement on one or more areas, the SOGIN, in agreement with the local authorities concerned and under the supervision of the ISIN, will carry out technical investigation campaigns in order to identify the site of the Deposito Nazionale. The ISIN control body must express its binding opinion with respect to the final confirmation of the suitability of the site. Subsequently, the Ministry of Ecological Transition, as established by Legislative Decree 31/2010, will identify the site with its own decree, which will also be issued if the various and repeated procedures for reaching the agreement fail.
According to the current programs, based on compliance with the deadlines dictated by current legislation, the works for the construction of the National Deposit and Technology Park will have a duration of about 4 years from the definitive identification of the area on which to build it. [129]

4.3.5 The Parco Tecnologico

As required by Legislative Decree 31/2010, together with the Deposito Nazionale, the Parco Tecnologico will be created, a research center on the decommissioning of nuclear plants and the management of radioactive waste, radiation protection and environmental protection. Research on the decommissioning and the management of radioactive waste will allow, for example, to develop new technologies to optimize the dismantling processes, improve the safety of operators and minimize the volumes of waste that will continue to be produced in our country. Research projects to promote the economic and industrial development of the area will also be agreed with the territory that will host the structure. The choice of the latter will be the subject of debate during the public consultation and will take

into account the vocations of the territories that will express their interest in hosting the Deposito Nazionale.
Depending on the activities of the Parco Tecnologico, two different financing models are envisaged. For research and development projects related to decommissioning activities and radioactive waste management, a minimum portion of the A2RIM component (formerly A2) of the electricity bill will be directly drawn, while for the activation of the other projects different sources of financing are assumed , both public and private.
The structures envisaged in the Parco Tecnologico project are the systems dedicated to activities related to the operation of the Deposito Nazionale and the laboratories to be managed in partnership with public or private entities that will allow, together with the employment and economic effects generated on the territory, to maximize the positive effects of the Parco Tecnologico.
The preliminary design of the Parco Tecnologico will be such as to allow to reduce the environmental impact of the infrastructure, through the use of solutions compatible with the ecosystem of the host territory. [130]

4.4 Safeguards and decommissioning technologies in Italy

As far as safeguards are concerned, according to the Law 332/03, the provisions of the AP are implemented by the Ministry for Economic Development, the Ministry for Foreign Affairs and the Ministry of Defence for military sites. On the basis of the agreement with the Ministry for Economic Development, the ENEA has the duty to perform verifications, studies, analyses and other specific activities related to the execution of the AP.
Equipments at the ENEA laboratories include the Small Samples Neutron Counter (SSNC), the In Situ Object Counting System (ISOCS), the INSPECTOR 1000 and the Sea Radioactive Waste Gamma Analyser (SRWGA). The SSNC is based on passive neutron counting: neutrons emitted by the sample are detected by $^{3}_{2}$He tubes and it provides the mass of $^{240}_{94}\mathrm{Pu}_{eq}$. The ISOCS is based on γ spectrometry with a Ge detector characterized by Monte Carlo code. The INSPECTOR 1000 is an instrument for both neutron and γ dosimetry: it is equiped with a $^{3}_{2}$He probe for neutrons detection and a LaBr detector for γ rays. The SRWGA is a system for the characterization of γ emitting materials (Figure 4.4). It implements different measurement techniques that allow the reconstruction of the distribution of the activity of the radionuclides in items containing radioactive materials. [131]
The SOGIN has a management dedicated to the study and implementation of projects related to the development of innovative technological solutions in nuclear decommissioning and radioactive waste management, with the objectives of improving safety and security, minimizing the production of waste, and increasing productivity, while reducing overall time and cost of activities.
The SOGIN uses, for radiological characterization activities, the Nucleco Integrated Waste Assay System (NIWAS) which includes various independent measurement techniques aimed at quantifying, with maximum precision and accuracy, the radiological content of a drum (Figure 4.5). In particular, the NIWAS system integrates the Nucleco Waste Assay System (NWAS) γ spectrometry based on the segmented scanning system, and the passive

Figure 4.4. The SRWGA. [131]

and active neutron count measurement using the Passive Active Neutron Waste Assay System (PANWAS) and the radiographic investigation. The results obtained with the listed techniques are integrated with the support of theoretical calculations and Monte Carlo simulations.

Figure 4.5. The NIWAS. [132]

The SOGIN is equipped with cutting-edge measurement systems such as the Tomographic Gamma Scanner (TGS), which allows you to reconstruct a 3D image of the matrix and of the contamination inside a drum and the Geomixed system is instead used for measurements on packages of variable geometries, for their unconditional removal.
In the ICS42 plant, the solid waste transferred to the Nucleco facilities is treated. After an initial repackaging and sampling phase for a complete physical and radiological characterization, such waste is sent to the Compaction and Dismantling Plant. In the ICS42 plant, the 200-liter drums containing the waste are reduced in volume through a 1500-ton hydraulic press and other equipment that allows for fully automated operations. After

compaction, the pellets produced are placed in special 400-liter metal containers, called Overpack, designed and approved for transport. The content of the Overpack is blanketed through a qualified and controlled cement mortar. The Overpacks are finally transferred to temporary deposits, pending the availability of a final disposal site.
The plant is also equipped with a Pretreatment and Dismantling section thanks to which it is possible to dismantle contaminated components of various sizes, inside dynamically and statically confined cells, shielded and equipped with automated and remote cutting equipment (torch plasma, manipulators, etc.). The Pretreatment and Dismantling section consists of a Pretreatment Cell and a Dismantling Cell. In the Pretreatment Cell, the volumetric reduction of solid waste takes place, with a 200-ton hydraulic press, and their packaging in 200-liter drums to be sent to the characterization, treatment and conditioning process. The Dismantling Cell, on the other hand, is dedicated to the volumetric reduction of large components by means of remote cutting with a plasma torch.
The SOGIN is designing a remote system that allows non-destructive tests to be performed, with particular attention to welding, on the outer shell of liquid radioactive waste storage tanks made of stainless steel. The system must be able to move autonomously in a confined environment with high dose rates and with the problem associated with the non-magnetic characteristics of the tanks.
The SOGIN has applied for a "utility model patent" to create an innovative treatment and conditioning system, by cementing, of a small volume of medium-activity liquid radioactive waste contaminated by plutonium, using devices and components installed inside a Glove Box. This system allows, in an α-sealed confined environment, to perform semi-automatic handling operations aimed at cementing this kind of waste and consists in the simplification and reduction of components and equipment normally used in large industrial plants (Figure 4.6). The system offers a solution to treat and condition small volumes of contaminated α aqueous liquid waste without having to build complex industrial plants that involve long authorization procedures, environmental impacts, significant construction and management costs.

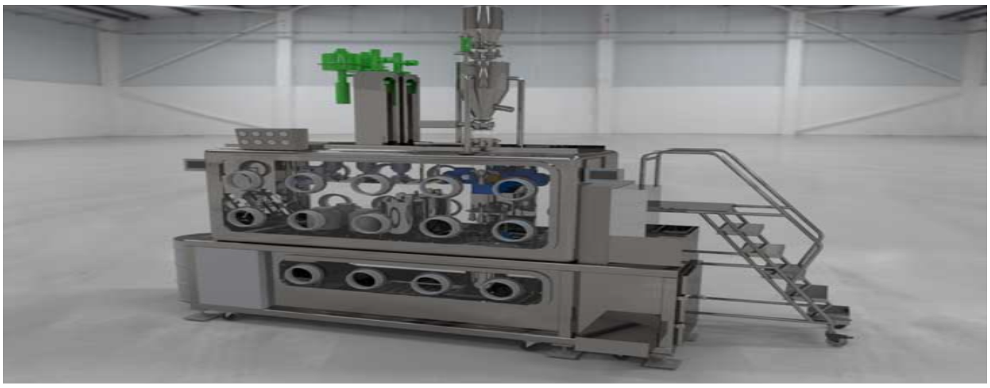

Figure 4.6. A Glove Box for cementing contaminated α liquid radioactive waste. [132]

The SOGIN is developing a Modular System for the Conditioning (Sistema di Condizonamento Modulare Rifiuti - SiCoMoR) of liquid waste. This prefabricated, modular and transportable plant has been designed to optimize its use by using it on multiple sites. Due to these characteristics, this system represents an overall management of radioactive waste with a reduced impact on the environment. The SiCoMoR currently under construction allows to solidify liquid radioactive waste through direct cementation in a cylindrical container. Its overall dimensions are about 20 meters long by about 17 meters wide. [132]

Chapter 5

Decommissioning and waste disposal with RESRAD

It is now clear that safe radioactive waste disposal is a very crucial and complicated task which must be faced with the right tools and approach. In areas where these wastes will be disposed of, or areas where contamination will not be fully remediated during clean-up, the two most important questions to ask are how much of the radiation left will eventually reach people and thus be able to cause them harm and what impact the radioactive materials will have on the larger ecosystem, including plants and other animals. Determining how humans may be affected by radioactive waste is a significant challenge and requires a great deal to be known about the properties of the site and how it may be used in the future. [133]

5.1 Why using RESRAD

Given that many of the necessary calculations are clearly quite complex, dose assessments are best done on a computer. [133] For this purpose, the Environmental Assessment Division of Argonne National Laboratory, United States (US) Department Of Energy (DOE) and US Nuclear Regulatory Commission (NRC), developed and then released in 1989 the RESRAD code. This program is used to evaluate radiological contaminated sites and to make regulatory decisions to help determine how clean the radioactivity levels at nuclear sites. RESRAD calculates radiological dose and allows users to specify the features of their site and to predict the dose received by an individual at time over the next 100,000 years. [134] Furthermore, it can be used successfully to identify major exposure pathways and estimate potential radiological risk to human health from contaminated soils when applied to real DOE sites using actual data. [135]
While there are other models, both from private and from government funded agencies, that have been developed to do similar types of calculations, RESRAD is particularly important for three reasons. First of all, RESRAD is the most extensively tested, verified, and validated code in the environmental risk assessment and site clean-up field. It has been accepted for use in making regulatory decisions and has been widely used by the DOE and its contractors. In 1994, the NRC approved the use of RESRAD for several

applications, including dose evaluation by licensees involved in decommissioning. The Environmental Protection Agency (EPA) used RESRAD in its rulemaking for clean-up of sites contaminated with radioactivity. Many industrial firms, universities, and foreign government agencies and institutions have used RESRAD, including the KORea RADioactive waste agency (KORAD), the Radioactive Waste Management Agency (RATA) and the Institute of Physics (headquartered in Lithuania) for the safety assessment for a near-surface disposal facility, [133] [134] [136] [137] [138] [139] and by the Ulsan National Institute of Science and Engineering for a safety assessment of the surrounding areas of Fukushima after the accident. [140] Second, RESRAD represents a robust, useful, and generally reasonable balance between the complexity of the problems involved with carrying out a dose assessment and the need for ease of usability and understandability of results. Third, and finally, RESRAD, unlike some other models, is easily available to any member of the public via the Internet so everyone can have equal access to the program and its supporting documentation. [133]

5.2 Main RESRAD characteristics

The basic framework of RESRAD has four major parts: source analysis, environmental transport analysis, dose/exposure analysis, and scenario analysis.
Source analysis addresses the source terms that determine the rate at which residual radioactivity is released into the environment. This rate is determined by the geometry of the contaminated zone, the concentrations of the radionuclides present, the ingrowth and decay rates of the radionuclides, and the removal rate by erosion and leaching.
Environmental transport analysis identifies the environmental pathways by which radionuclides can migrate from the source to a human exposure location and determines the migration rate along these pathways. There are three main exposure pathways, which are external radiation, inhalation and ingestion. For each of them, radionuclides can migrate from a source to a human exposure location by many environmental pathways, and RESRAD models nine of them.
External γ radiation from radionuclides distributed throughout the soil is the dominant external radiation pathway and the only external radiation pathway taken into account.
Ingestion pathways consist of food, water, and soil ingestion pathways. Four food pathway categories are considered: plant foods, meat, milk, and aquatic foods. Water independent and water dependent pathways are included. The water independent one includes the following four plant food pathways: root uptake from crops grown in the contaminated zone, foliar uptake from contaminated dust deposited on the foliage, root uptake from contaminated irrigation water, and foliar uptake from contaminated irrigation water. The water dependent ones include surface and well water. Both well water and surface water can be used for drinking. The fraction of well water blended with or supplemented by surface water is used to calculate the total contribution from groundwater and surface water. The ingestion pathway also includes direct ingestion of contaminated soil itself.
Inhalation exposure results primarily from inhalation of contaminated dusts. Radon has its own separated environmental pathway, both the water dependent and the water independent.
A scheme of all these pathways modeled by RESRAD are shown in Figure 5.1.

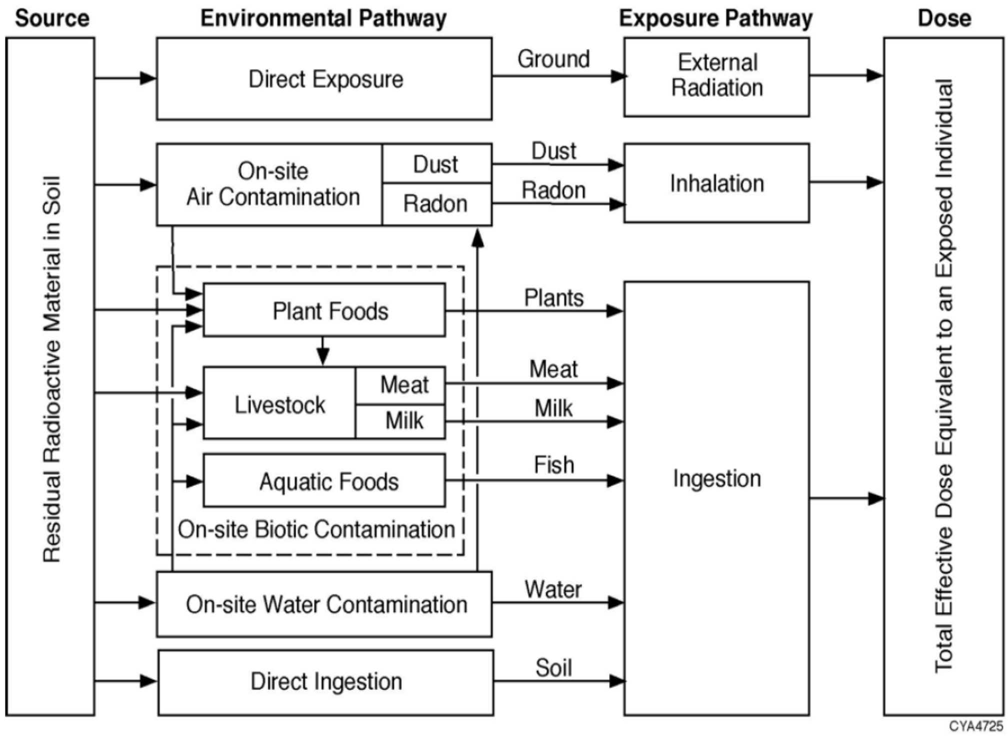

Figure 5.1. A schematic representation of RESRAD pathways. [136]

RESRAD calculates radionuclide concentrations in different environmental media as a function of time. The details of the models implemented are illustrated in the guides. [133] [136] [141] RESRAD then uses the concentrations in different environmental media to compute intake, dose, and risk. The total dose or risk is the sum of dose or risk from individual pathways. The radiological doses and risks are all time-integrated quantities. The dose reported for a particular time is the dose over a period of one year beginning at the specified time. The risk reported for a particular time is the value over an exposure duration beginning at the specified time. The exposure duration for risk calculation is the length of time the receptor is exposed to radiation at the site. Hence, exposure duration may be different for different exposure scenarios. RESRAD finds the time-integrated dose, or risk, by performing a trapezoidal integration using the dose, or risk, rates at the beginning and end of the time interval and at all the calculation time points that fall within the time interval (one year for dose and the exposure duration for risk). If the end of the time integration interval does not coincide with a calculation time point, the rate at the end is found by interpolating between the two calculation time points around it. The total annual dose, $D(t)$, received by a member of the critical population group at time t following the radiological survey of the site ($\frac{mSv}{y}$) is given by 5.1.

$$D(t) = \sum_{i=1,n} \sum_{p} D_{ip}(t) \tag{5.1}$$

where $D_{ip}(t)$ is the annual dose received by a member of the critical population group beginning at time t from the i^{th} principal radionuclide transported through the p^{th} environmental pathway, together with its associated decay products ($\frac{mSv}{y}$), $D(t)$ is the sum of annual doses over all active pathways, p, and the number of principal radionuclides present, n.
Principal radionuclides are radionuclides with half-lives greater than the cut-off half-life selected. In RESRAD the user can select any cut-off half-life greater than or equal to ten minutes. The decay products of any principal radionuclide down to, but not including, the next principal radionuclide in its decay chain are called associated radionuclides. The code assumes that the associated radionuclides are in secular equilibrium with the preceding principal radionuclide during transport and at the point of exposure.
The criterion for releasing a site for use without radiological restrictions is generally based on dose limit. In Italy the limits, which are imposed by the Legislative Decree 230/95 and subsequent amendments, for the exposed workers and the population correspond to those recommended internationally by the International Commission on Radiological Protection (ICRP), which are 20 $\frac{mSv}{y}$ for the exposed workers belonging to Category A, 6 $\frac{mSv}{y}$ for the exposed workers belonging to Category B, and 1 $\frac{mSv}{y}$ for the general population. [141]
Even though RESRAD is a powerful tool, it has some limitations. While a great many exposure pathways are accounted for in the program, RESRAD cannot currently predict doses to the embryo/fetus or to a breast fed infant nor can it predict doses arising from swimming in contaminated water. Dose conversion factors relating a woman's intake of radionuclides to the dose received by the embryo/fetus or breast fed infant were published by the ICRP in 2002 and 2004. However, these dose conversion factors cannot be used directly in RESRAD because there are not adequate input parameters in the model to account for the exposure of the embryo/fetus or breast fed infant resulting from maternal pathways. RESRAD cannot correctly calculate doses to children from external irradiation. Doses from external radiation are calculated using the EPA's Federal Guidance Report 12 where the doses are calculated for the average of a 59 kilogram, 160 centimeter tall woman and a 70 kilogram, 170 centimeter tall man. [133]

5.3 Reference case

In order to show the reliability of the code, the results of an already performed radiological safety assessment [139] are reproduced by using RESRAD-ONSITE.
Figure 5.2 shows the interface of RESRAD.

5.3.1 Input data and scenarios

RESRAD requires a lot of input data to perform a correct radiological assessment. The radionuclides analysed in the reference article and their initial concentrations are shown in Table 5.1, while Table 5.2 shows the input parameters. All the others are set to their default value.

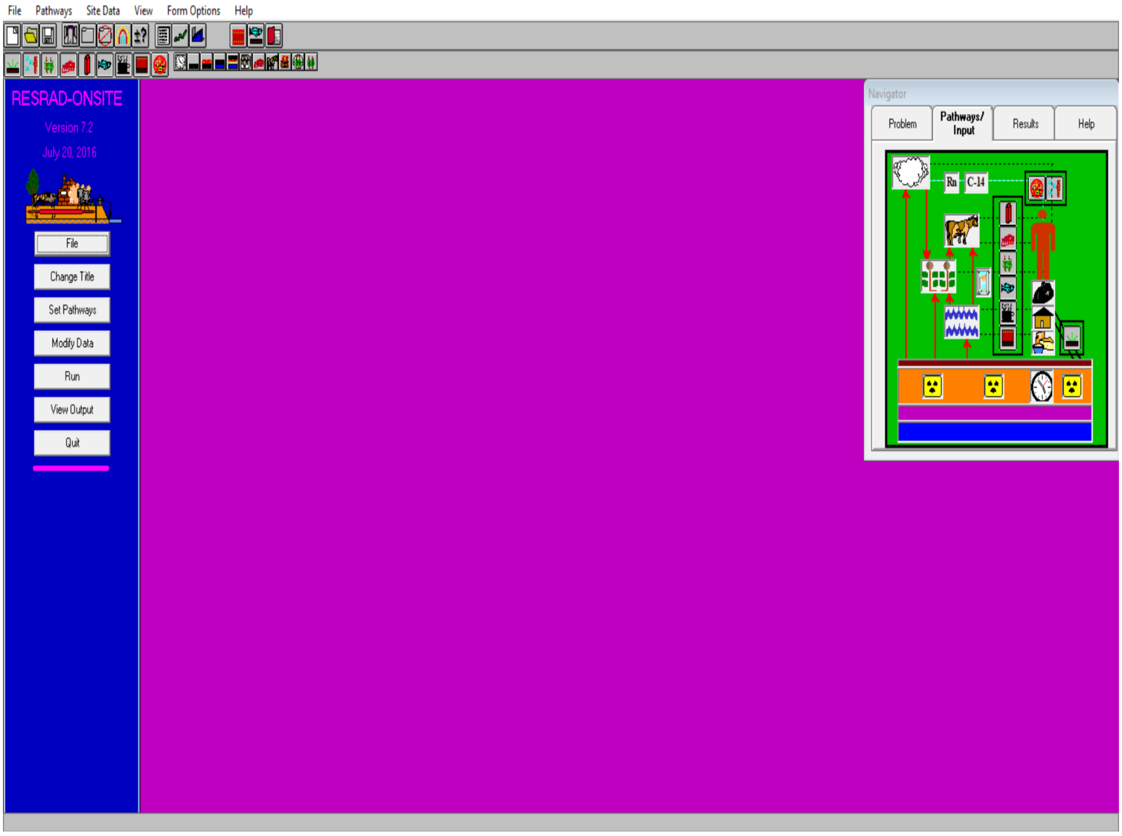

Figure 5.2. A representation of RESRAD interface.

Radionuclide	**Soil concentration** $[\frac{Bq}{g}]$
Ce-144	$5.16 * 10^{-3}$
Co-57	4.41
Co-60	7.51
Cs-137	1.03
Fe-55	$2.96 * 10^{1}$
Nb-94	$6.1 * 10^{-4}$
Ni-63	$1.83 * 10^{1}$
Sr-90	$2.54 * 10^{-2}$

Table 5.1. Initial soil concentrations. [139]

Exposure scenarios are patterns of human activity that can affect the release of radioactivity from the contaminated zone and the amount of exposure received at the exposure location. The four principal scenarios are rural resident farmer, urban resident, worker, and recreationist. The principal exposure scenarios can give rise to specific sub-scenarios. The exposure scenarios that need to be considered for each site depend on the potential

Parameter	**Value**
Area of the contaminated zone	2500 m^2
Thickness of the contaminated zone	0.15 m
Cover depth	1 m
Density of the cover material	1.5 $\frac{g}{cm^3}$
Cover erosion rate	0.001 $\frac{m}{y}$
Contaminated zone erosion rate	0 $\frac{m}{y}$
Precipitation	1 $\frac{m}{y}$
Wind speed	2 $\frac{m}{s}$
Inhalation rate	8,400 $\frac{m^3}{y}$
Mass loading for inhalation	0.0001 $\frac{g}{m^3}$
Indoor time fraction	0.5
Outdoor time fraction	0.25
Fruit, vegetable and grain consumption	190 $\frac{kg}{y}$
Leafy vegetable consumption	100 $\frac{kg}{y}$
Milk consumption	63 $\frac{l}{y}$
Meat and poultry consumption	55 $\frac{kg}{y}$
Fish consumption	79.3 $\frac{kg}{y}$
Other seafood consumption	33.4 $\frac{kg}{y}$
Soil ingestion	36.5 $\frac{g}{y}$
Drinking water intake	196.3 $\frac{l}{y}$
Fruits, non-leafy vegetables and grain storage time	14 d
Leafy vegetables storage time	1 d
Milk storage time	1 d
Meat storage time	7 d
Fish time	1 d
Crustacea and mollusks storage time	1 d
Well water storage time	0.5 d
Surface water storage time	0.5 d
Livestock fodder storage time	75 d

Table 5.2. Input parameter values used. [139]

land use of that specific site and then the exposure groups of receptors that are relevant to each of those potential land uses must be considered. Different combinations of land use and exposure group can lead to the same generic exposure scenario and they would each have different input parameters to reflect the differences in land use and in activities these people perform. Thus, the land use/exposure group approach is necessary for conducting site-specific analyses and developing input values for the exposure scenarios. [136] [142] Table 5.3 shows which pathways are accounted for in the main different scenarios.

Pathway	Resident farmer	Suburban resident	Industrial worker	Recreationist
External radiation	Yes	Yes	Yes	Yes
Inhalation	Yes	Yes	Yes	Yes
Radon	Yes	Yes	Yes	Yes
Plant ingestion	Yes	Yes	No	No
Meat ingestion	Yes	No	No	Yes
Milk ingestion	Yes	No	No	No
Aquatic foods	Yes	No	No	Yes
Soil ingestion	Yes	Yes	Yes	Yes
Drinking water	Yes	No	No	No

Table 5.3. Pathways considered in the main scenarios. [136]

Among the various possible scenarios, two of them have been selected for this case study. The residential farmer scenario has been chosen because it is the most conservative since it takes into account all possible exposure pathways. However, an actual scenario for a specific site should be determined taking into account many related factors such as location, use, scope and physical characteristics of the site. Considering this, a more realistic scenario is the industrial worker one.

5.3.2 Results

The total dose has been calculated to understand the impact on the population, in particular if it overcomes the dose limit for the public. In Figure 5.3 it is shown the total dose together with the contribute from each radionuclide obtained in the residential farmer scenario, compared with the results of the reference article. In particular, the continuous lines represent the results of the simulation, the dashed lines represent the results from the article.
It can be seen that the two trends are very similar. In particular, the total exposure dose decreases with time thanks to the radioactive decay and the influence of natural phenomena. However, at a certain point the total dose started to rise with time. This can be due to the fact that radionuclides adsorbed in soil were leached by infiltrating water from the contaminated zone and reached to groundwater used by the public. After the peak, it again decreases but it can be noticed that in any case the public dose limit is never reached. The graph shows also that the radionuclides that contribute the most are ${}^{60}_{27}Co$, ${}^{63}_{28}Ni$ and ${}^{137}_{55}Cs$.
This trend is also reflected in the excess cancer risk curve, as seen in Figure 5.4.

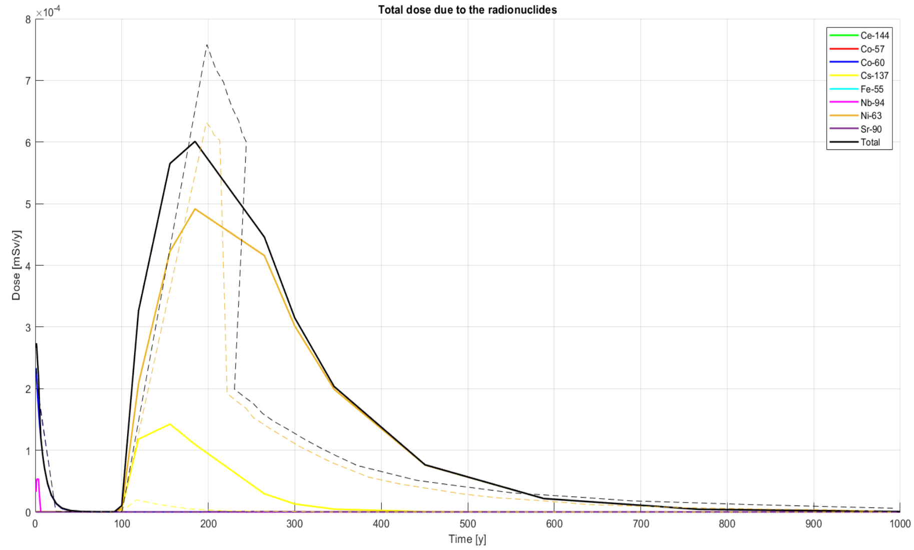

Figure 5.3. The exposure dose from each radionuclide.

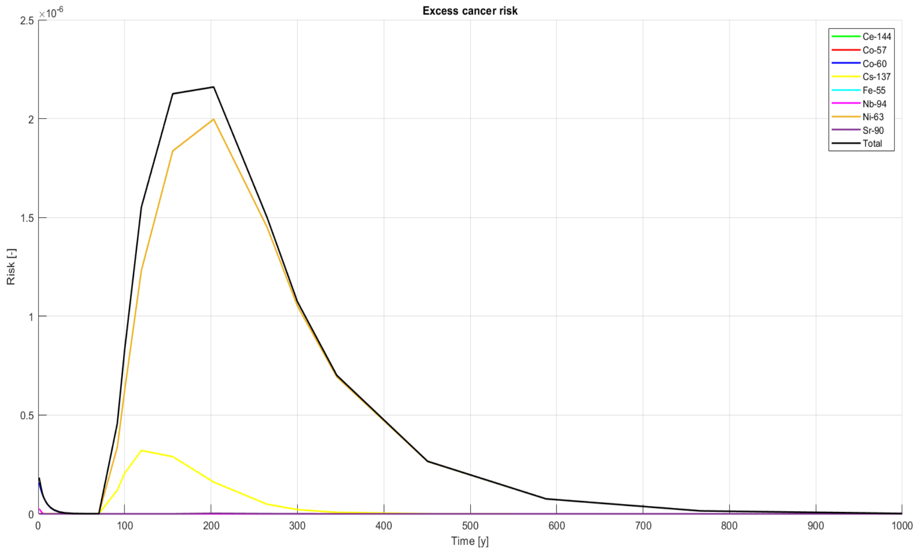

Figure 5.4. The excess cancer risk from each radionuclide.

Another important aspect is the contribution of each pathway. Figure 5.5 shows the comparison between the results obtained and the ones from the reference article.
Again the two trends are quite close together. At the beginning the most important

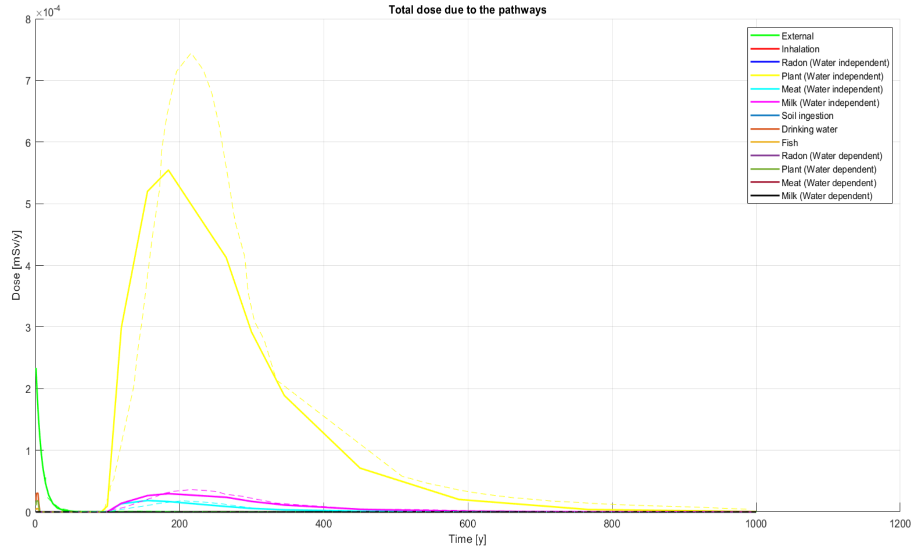

Figure 5.5. The exposure dose form each pathway.

pathway is the external radiation but then the main contribution comes from the water-independent plant ingestion.
In Figure 5.6 it can be seen that this is true also for the excess cancer risk.

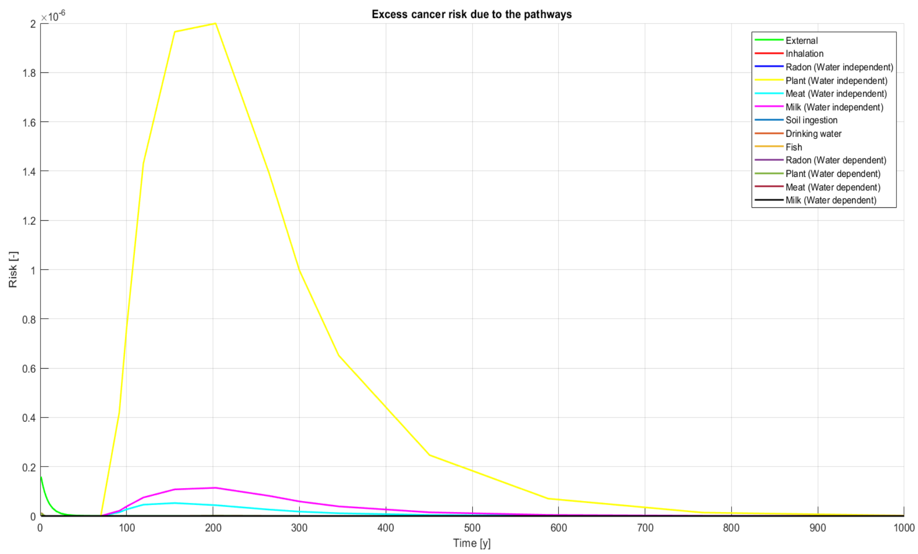

Figure 5.6. The excess cancer risk from each radionuclide.

Then, an industrial worker scenario simulation has been performed and its total dose has been compared to the residential farmer scenario one in Figure 5.7.

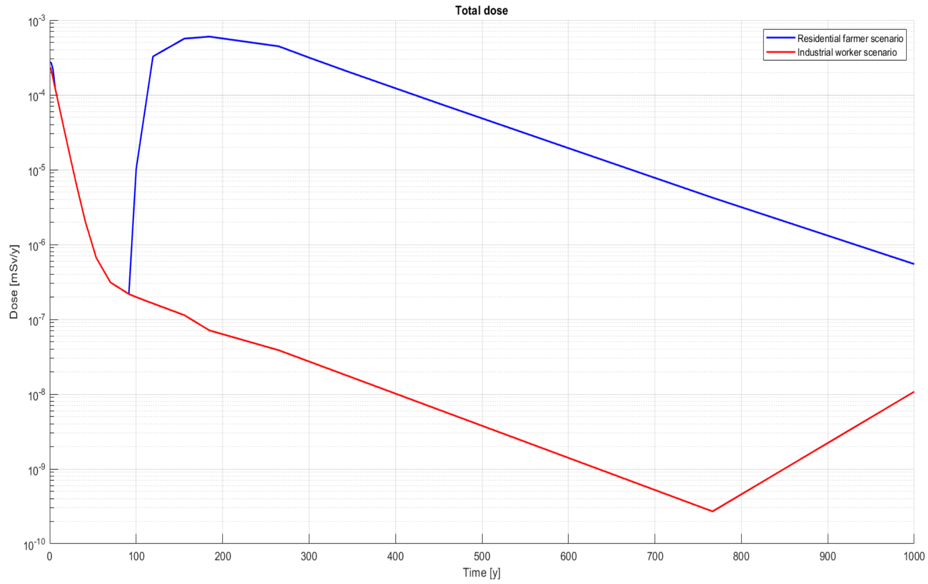

Figure 5.7. The total exposure dose in the two scenarios.

At the beginning there are no differences because of the dominance of the external radiation pathway but after, due to the other pathways presence, the residential farmer scenario dose becomes grater. The industrial worker scenario does not have the same increase of the residential farmer one due to the latter big contribution of the drinking water and the contaminated groundwater used by the population. In the industrial worker scenario there is an increase towards the end which is due to the soil pathway of $^{63}_{28}Ni$. [139]

5.3.3 Sensitivity analysis

A sensitivity analysis has been carried out to see the influence of some important input data over the total dose.
An important parameter that affects the final result is the cover depth: it corresponds to the distance from the surface to the uppermost contaminated soil sample. Thus, it is important to have enough distance in order to ensure the right protection from the radionuclides. To analyse its importance, three different simulations with three different values of the cover depth have been performed and the results are shown in Figure 5.8.
As the cover depth decreases, the total dose increases, and if the cover depth reaches 0.1 m it also overcomes the public limit. This aspect underlines the importance of this parameter. This behaviour occurs also in the reference article.
Not only the depth but also the density of the covering material must be rightly chosen to protect the people and the environment. This is why three values of the material density

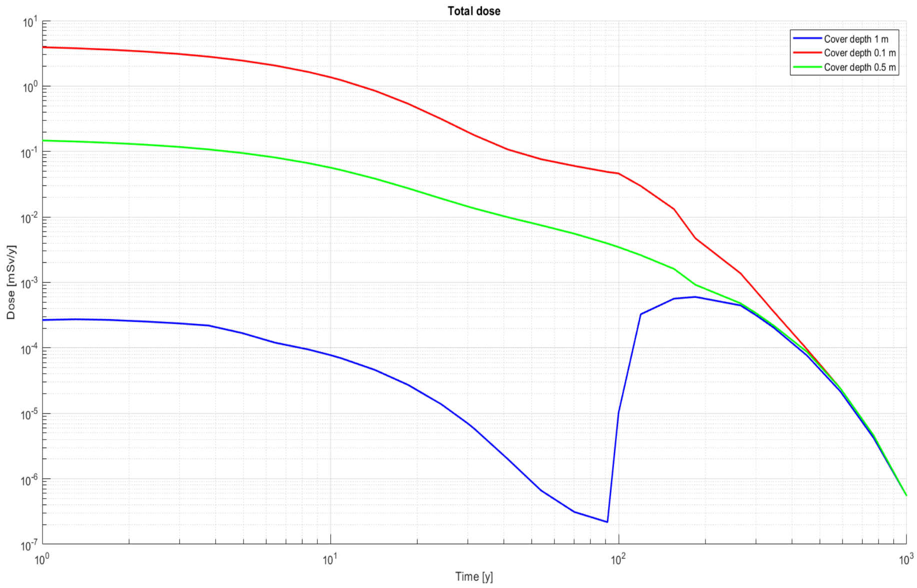

Figure 5.8. The total exposure dose varying the cover depth.

where chosen and its effects are shown in Figure 5.9.

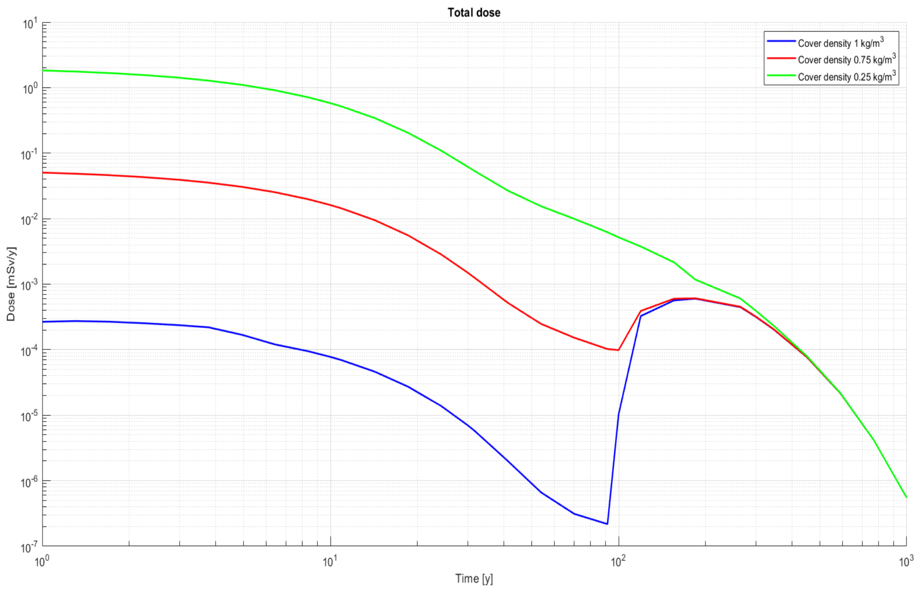

Figure 5.9. The total exposure dose varying the cover density.

As in the reference article, by decreasing the material density, the total dose increases but its impact is lower with respect to the cover depth.[139]

Chapter 6

Italian case study

Securing the safety of the radioactive waste repository is an important question, due to the potential radiation hazard from the disposed radionuclides. Therefore, understanding of radionuclides behavior in the natural environment is of critical importance for a reliable safety assessment of radioactive waste repositories. [138] It has been already mentioned in the second chapter that, among the various decommissioning wastes, LLW should be disposed of into near-surface disposal facilities. Therefore, radiological impact analyses and evaluation of doses to population is necessary, starting from the foreseen radioactive waste repository operation plan.
This is why in this thesis RESRAD-ONSITE code has been used for a case study suitable for the Italian situation.

6.1 Waste Acceptance Criteria

One of the main criteria for ensuring the safety of the Deposito Nazionale is the use of the Waste Acceptance Criteria (WAC): radioactive waste will be accepted only if its characteristics will comply with indications and technical-managerial prescriptions. The WACs therefore represent a measure of the performance required of the artifacts of radioactive waste in order to ensure the safety of operators, of the people and of the environment during all stages of the life of the deposit and during all scenarios. [143]
The WACs cover all aspects that play a role in management of radioactive waste. The characteristics of the waste that may cause radiological risks to humans and the environment are considered. In addition to the radiological aspects, the characteristics of the materials that can favor processes degradation inside the building (generation of gas, heat, corrosion, swelling, accumulation of secondary products), and therefore compromise efficiency of the conditioning matrix or container, favoring the release of radionuclides, are considered. Furthermore, the properties of the waste-form, that allow to produce a stable product, the characteristics of the containers, that guarantee the radioactivity containment function and the management of the artifact until its insertion in the module (type of material, geometry, dimensions, processes of potential degradation), and the characteristics related to the waste packaging process (specific activity, total activity, fissile material, dose rate, surface contamination), are considered.

The WACs are generally derived from specific safety analyses that are used to verify the behaviour of the waste, of the waste form and of the product under certain conditions.
The WACs for LLW and ILW are developed for their acceptability to the Deposito Nazionale. Radioactive waste carries a radiological risk which depends on the type and on the amount of radionuclides contained. Furthermore, radioactive waste must not contain materials that could lead to accelerate degradation of engineering barriers in the short and long term. The possible mixing of materials of different origins and types that may have properties incompatible from the chemical/physical viewpoint, such as to trigger similar degradation processes, must be also evaluated in advance. For example, materials that cannot be accepted are liquid waste, explosive, flammable, strongly reactive and putrescible materials. Some materials are generally acceptable with limitations in order to reduce the risk associated with their specific characteristics. This includes liquids that are not bounded to the conditioning matrix, absorbing, soluble, organic materials and sealed sources.
The total activity present in the artifact must be uniformly distributed inside it, in order to avoid the creation of hot spots inside the volume. It must have an adequate weight to be safely handled, during transport activities and in all life stages of the Deposito Nazionale. The amount of fissile material in each artifact will have to be limited in order to exclude that critical events may occur. Removable surface contamination must be less than 4 $\frac{Bq}{cm^2}$ for β and γ emitting radionuclides, and less than 0.4 $\frac{Bq}{cm^2}$ for α emitting radionuclides. The dose rate must be less than 2 $\frac{mSv}{h}$ in contact with the artifact, and less than 0.1 $\frac{mSv}{h}$ 1 m far from the external surface of the artifact.
The artifact must be characterized by a radiological content such as to ensure compliance with the dose limits prescribed for operators and the population in the different stages of the life of the Deposito Nazionale. This criterion translates into limits on the maximum concentration of assets for each radionuclide, expressed in terms of $\frac{Bq}{g}$, derived from the safety analyses carried out in the different scenarios. This criterion can be defined downstream of the final safety assessment that will be performed on the site selected for the realisation of the Deposito Nazionale. In the absence of the definitive safety assessment, it is possible to take as preliminary reference the activity concentration values indicated in the radioactive waste classification (DM 7^{th} August 2015) for the "Low Activity" category. [120] [144]
By means of the WACs it is possible to define the source term. It is the quantity of radioactive material that is authorized for disposal in compliance with dose objectives. It is essentially characterised by short-lived radionuclides (half-life equal to or less than 31 years) and limited quantities of long-lived radionuclides (400 $\frac{Bq}{g}$) such that it allows reaching, within a few hundred years, concentrations of radioactivity that do not involve impacts for health and the environment. If the design features of the Deposito Nazionale and the geological characteristics of the site do not guarantee compliance of the dose targets for the entire inventory destined for disposal, it will be necessary to operate a reduction of the the radioactive inventory. [143]

6.2 Repository calculations

Two types of calculations have been performed with RESRAD. Firstly, the maximum allowable concentrations of each radionuclide of Table 5.1 have been calculated. Secondly, by supposing that the abovementioned inventory concentration is made of the same radionuclides in the same proportion of the soil concentration of Table 5.1, the dose exposure has been calculated.
Table 6.1 shows the activity of the Italian LLW inventory while Table 6.2 shows the volume of this inventory. These data are updated to December 2019.

Facility	**Activity** $[GBq]$
Caorso	2,187.21
Garigliano	20,782.45
Latina	16,444.18
Trino	896.75
Eurex	255.68
Itrec	2,997.63
OPEC 1	57.42
Plutonium plant	0
Bosco Marengo	34.36
Nucleco	3115.28
JRC Ispra	493.82
Avogadro	403.95
Others	210.07
Total	47,878.8

Table 6.1. Activity inventory of the LLW stored in Italy up to December 2019. [145]

Facility	**Volume** $[m^3]$
Caorso	1,584.9
Garigliano	1,149.66
Latina	489.1
Trino	201.48
Eurex	890.94
Itrec	356.28
OPEC 1	2.79
Plutonium plant	0
Bosco Marengo	329.41
Nucleco	3,734.1
JRC Ispra	3,235.3
Avogadro	3.22
Others	544.01
Total	12,521.19

Table 6.2. Volume inventory of the LLW stored in Italy up to December 2019. [145]

6.2.1 Maximum concentrations

In order to calculate the maximum allowable concentrations of each radionuclide, it is assumed that the total waste inventory is made only of one radionuclide and then, through the radionuclide density, the specific activity of the radionuclide under investigation is obtained and finally, according to the results, its concentration is increased or decreased up to the point that it causes a 1 $\frac{mSv}{y}$ dose.
The first step consists of calculating the dose from a unitary concentration of each radionuclide. The results are shown in Table 6.3.

Radionuclide	**Maximum dose** $[\frac{mSv}{y}]$
Ce-144	$1.687 * 10^{-7}$
Co-57	$1.089 * 10^{-12}$
Co-60	$3.512 * 10^{-5}$
Cs-137	$1.442 * 10^{-4}$
Fe-55	$1.081 * 10^{-18}$
Nb-94	$8.757 * 10^{-2}$
Ni-63	$2.765 * 10^{-5}$
Sr-90	$6.476 * 10^{-5}$

Table 6.3. Dose of each radionuclide with unitary initial concentration.

In order to get the concentrations, it is necessary to know the densities of the radionuclides. They are shown in Table 6.4.

Radionuclide	Density $[\frac{g}{m^3}]$
Ce-144	$6.76 * 10^6$
Co-57	$8.9 * 10^6$
Co-60	$8.9 * 10^6$
Cs-137	$1.9 * 10^6$
Fe-55	$7.8 * 10^6$
Nb-94	$8.4 * 10^6$
Ni-63	$8.9 * 10^6$
Sr-90	$2.6 * 10^6$

Table 6.4. Density of each radionuclide. [146]

Now it is possible to get the specific activity of each radionuclide and the corresponding dose if the entire Italian LLW inventory were made only of the radionuclide under consideration. They are shown in Table 6.5.

Radionuclide	Specific activity $[\frac{Bq}{g}]$	Maximum dose $[\frac{mSv}{y}]$
Ce-144	565.65	$9.55 * 10^{-5}$
Co-57	429.64	$4.68 * 10^{-10}$
Co-60	429.64	$1.51 * 10^{-2}$
Cs-137	2012.54	$2.9 * 10^{-1}$
Fe-55	490.23	$5.3 * 10^{-16}$
Nb-94	455.22	$4 * 10^{1}$
Ni-63	429.64	$1.19 * 10^{-2}$
Sr-90	1470.7	$9.52 * 10^{-2}$

Table 6.5. Specific activity of each radionuclide.

Finally, thanks to these values, the maximum allowable concentrations are reported in Table 6.6, together with the maximum dose reached. The doses trends are also shown in Figures 6.1-6.2-6.3-6.4-6.5-6.6-6.7-6.8.

Radionuclide	**Maximum concentration** $[\frac{Bq}{g}]$	**Maximum dose** $[\frac{mSv}{y}]$
Ce-144	$5 * 10^6$	$8.44 * 10^{-1}$
Co-57	$9 * 10^{11}$	$9.8 * 10^{-1}$
Co-60	$2.5 * 10^4$	$8.78 * 10^{-1}$
Cs-137	$6 * 10^3$	$8.65 * 10^{-1}$
Fe-55	$9 * 10^{17}$	$9.73 * 10^{-1}$
Nb-94	$1 * 10^1$	$8.76 * 10^{-1}$
Ni-63	$3 * 10^4$	$8.29 * 10^{-1}$
Sr-90	$1.5 * 10^4$	$9.71 * 10^{-1}$

Table 6.6. Maximum concentration of each radionuclide.

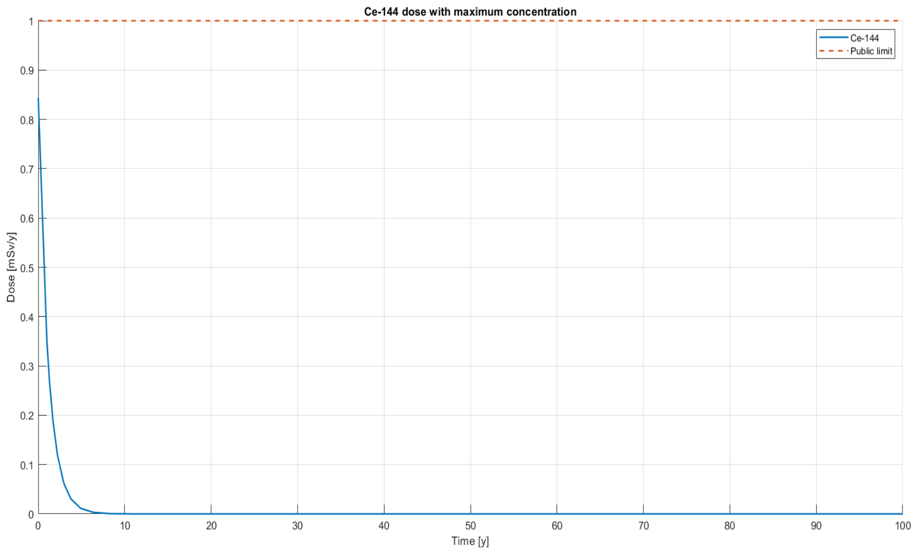

Figure 6.1. The exposure dose from Ce-144 when it has the maximum concentration.

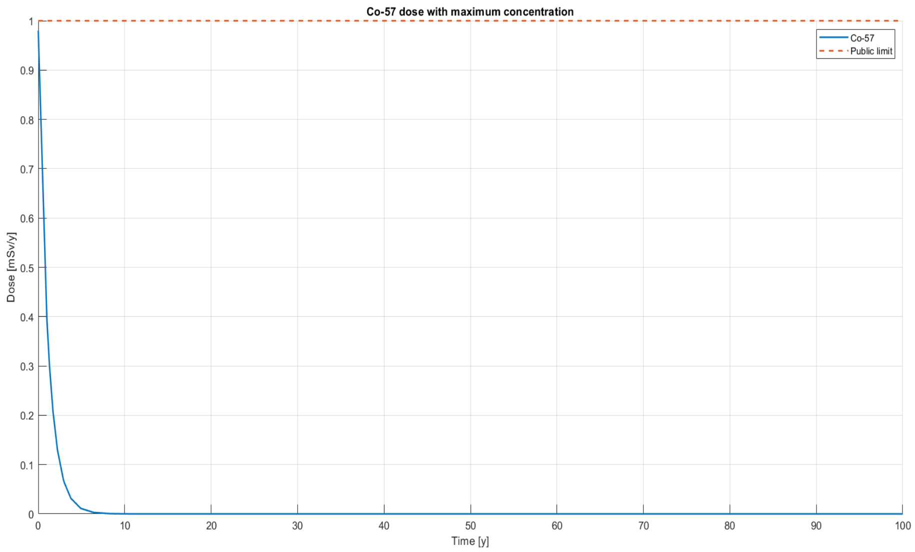

Figure 6.2. The exposure dose from Co-57 when it has the maximum concentration.

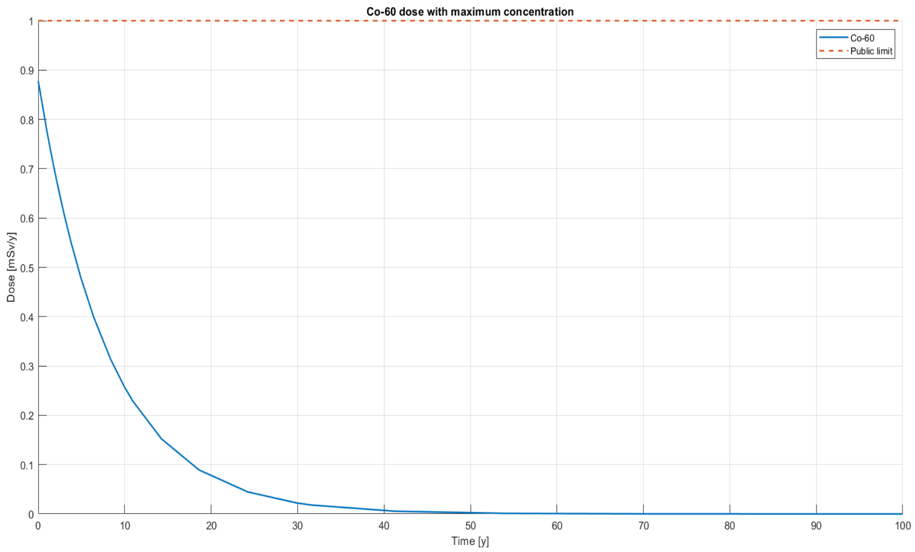

Figure 6.3. The exposure dose from Co-60 when it has the maximum concentration.

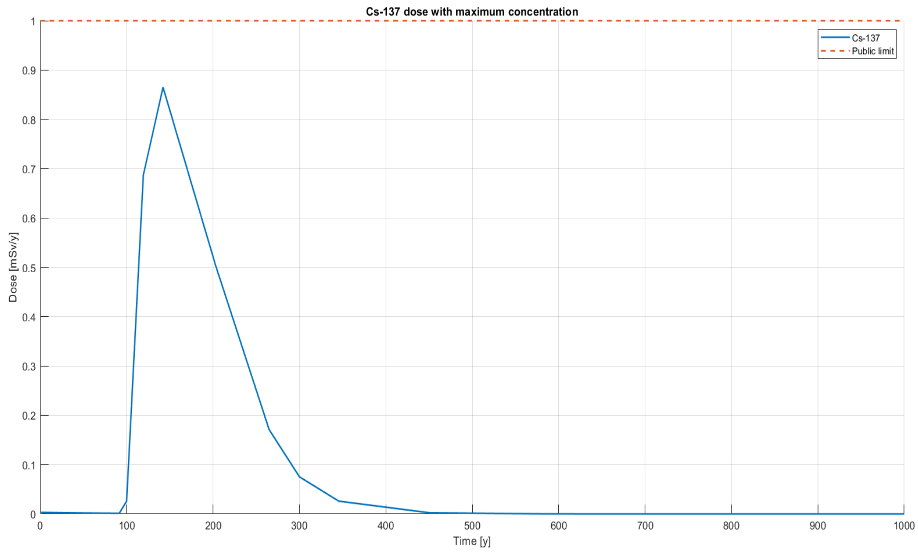

Figure 6.4. The exposure dose from Cs-137 when it has the maximum concentration.

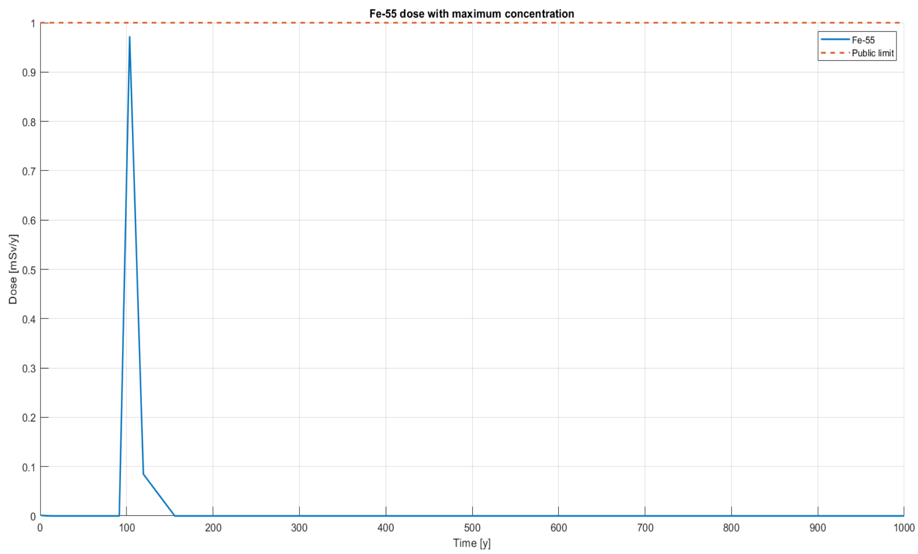

Figure 6.5. The exposure dose from Fe-55 when it has the maximum concentration.

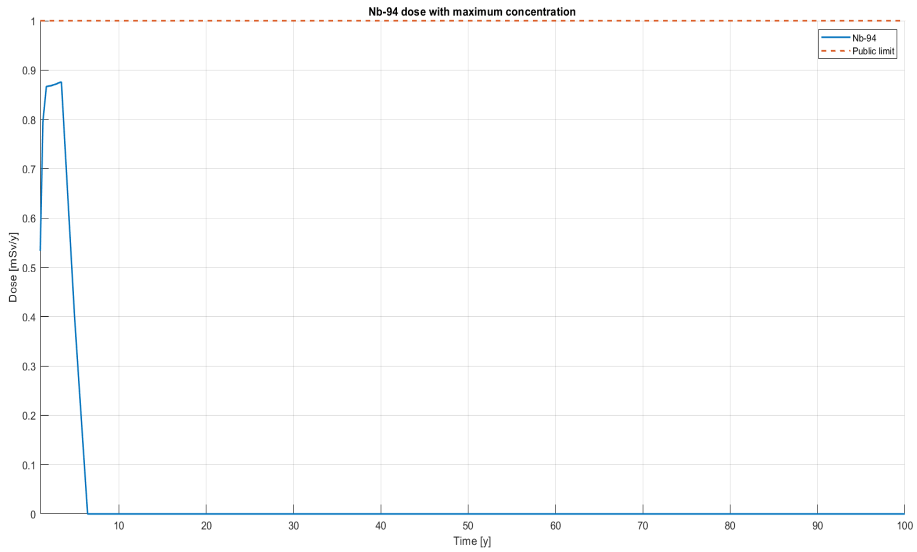

Figure 6.6. The exposure dose from Nb-94 when it has the maximum concentration.

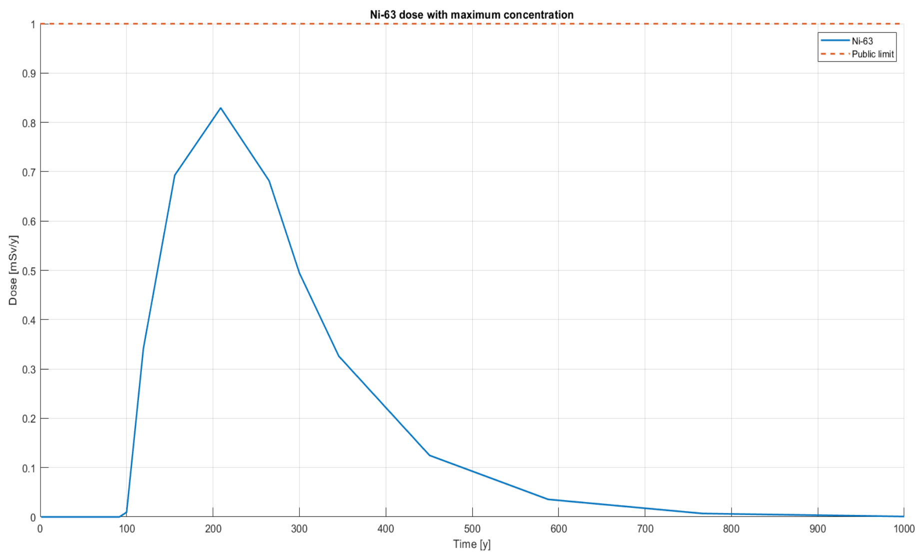

Figure 6.7. The exposure dose from Ni-63 when it has the maximum concentration.

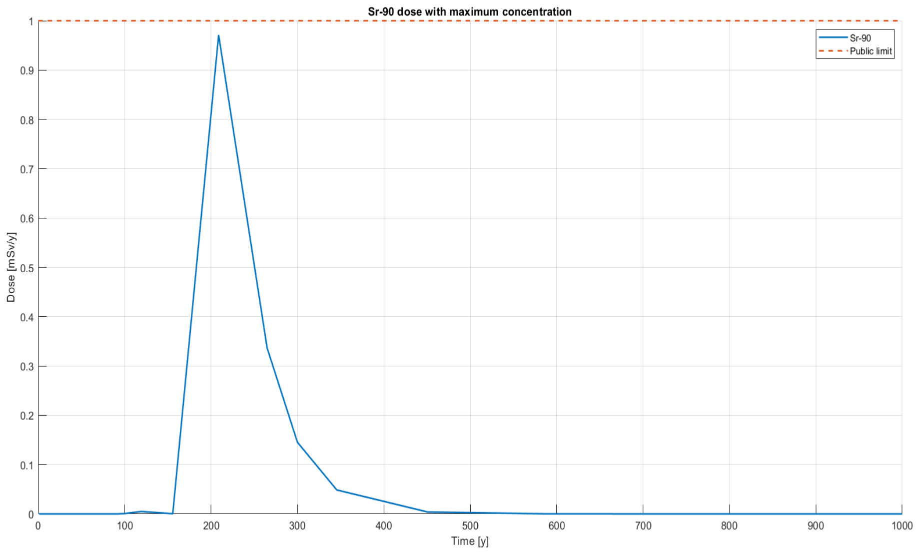

Figure 6.8. The exposure dose from Sr-90 when it has the maximum concentration.

6.2.2 Repository dose

In order to calculate a reference dose for the Deposito Nazionale, it is assumed that the radionuclides in the Italian LLW inventory are distributed in the same way as the waste from the reference article. [139] Table 6.7 shows the percentages of each radionuclide.

Radionuclide	**Soil concentration** $[\frac{Bq}{g}]$	**Percentage**
Ce-144	$5.16 * 10^{-3}$	0.008%
Co-57	4.41	7.244%
Co-60	7.51	12.336%
Cs-137	1.03	1.692%
Fe-55	$2.96 * 10^{1}$	48.619%
Nb-94	$6.1 * 10^{-4}$	0.001%
Ni-63	$1.83 * 10^{1}$	30.058%
Sr-90	$2.54 * 10^{-2}$	0.042%

Table 6.7. Radionuclides percentages.

Since most of these radionuclides are steel activation products, when converting from total activity to specific activity, the steel density has been used, which is $7.9 * 10^{6}$ $\frac{g}{m^3}$[147]. Thus, the overall specific activity comes out to be 484.028 $\frac{Bq}{g}$.
Now it is possible to calculate the radionuclides concentrations. They are reported in Table 6.8.

Radionuclide	**Soil concentration** $[\frac{Bq}{g}]$
Ce-144	$3.87 * 10^{-2}$
Co-57	$3.51 * 10^{1}$
Co-60	$5.97 * 10^{1}$
Cs-137	8.19
Fe-55	$2.35 * 10^{2}$
Nb-94	$4.84 * 10^{-3}$
Ni-63	$1.45 * 10^{2}$
Sr-90	$2.03 * 10^{-1}$

Table 6.8. Reference initial soil concentrations.

Finally, the dose is calculated with RESRAD. The results are shown in Figure 6.9.
As it can be seen, it is well below the public dose limit: according to our estimates, with the reference concentrations for LLW and the Deposito Nazionale configuration, doses to public never exceed 5 $\frac{\mu Sv}{y}$ for a limited time span, peaking before 200 years after disposal. Such dose levels, from the radiological viewpoint, are negligible.

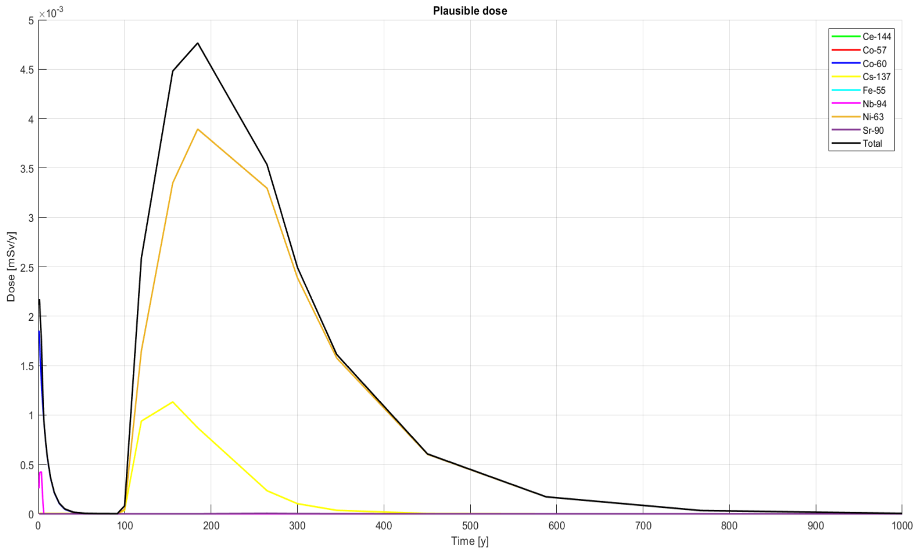

Figure 6.9. A reference dose from an Italian repository.

Chapter 7

Conclusions

This thesis explains the main characteristics of nuclear safeguards and decommissioning and how a first safety assessment is carried out.
The IAEA can rely on an adequate amount of tools in order to fight nuclear proliferation. Starting with the basic nuclear accountancy and inspections, the IAEA can furtherly check the nuclear material by means of seals and surveillance techniques, together with sampling of the environment and the correct verification of the information provided by the States. The strength of the safeguards system changes according to the type of agreement stipulated between the State and the IAEA and every non-compliance is reported.
Nuclear decommissioning is a long and complex activity, but it is fundamental for the sustainability of nuclear energy. Thanks to it, radioactive waste can be safely disposed (according to their classification) and nuclear plants can be safely dismantled, allowing a reuse of the site.
From the point of view of security, safeguards must be applied also during nuclear decommissioning. By means of DA and NDA techniques, spent nuclear fuel can be measured and so it is possible to verify any possible misuse or diversion scenario. Furthermore, during the decommissioning process, annual verification of information and inventory are performed.
Italy has no new nuclear facilities under construction, but it needs to dismantle its old fleet made of four nuclear plants, five facilities related to the nuclear fuel cycle and a research reactor. The construction of the Deposito Nazionale, for the disposal of VLLW and LLW and the temporary storage of ILW and HLW, is ongoing: this year, on 5^{th} January, the CNAPI was published and the subsequent public consultation started. Following that, on 7^{th} September the National Seminar started and it will end on 15^{th} December, and after all this, the SOGIN will publish the CNAI.
An important aspect that arises during the decommissioning is ensuring that no harm occurs to the population and the environment. Thus, the use of a software tool able to predict the dose and the risk is crucial. A program able to provide accurate results is RESRAD: by specifying the site characteristics, this code calculates radionuclides concentrations as a function of time and it uses them to calculate the radiological doses and risks.
This code has been used in this thesis to carry out a first radiological assessment of a hypothetical near-surface deposit. First of all, a simulation of a reference situation has been performed to show that RESRAD is a reliable code. Indeed, the results are very similar to some results found in literature. Finally, simulations more related to the Italian

case have been carried out: firstly, the maximum allowable concentrations of the main radionuclides in the LLW inventory are calculated, and then the total dose to population: the latter is well below the public dose limit: according to our estimates, doses to public never exceed 5 $\frac{\mu Sv}{y}$ for a limited time span. Such dose levels are, from the radiological viewpoint, negligible.

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