A Comparative Analysis
of Decommissioning Scenarios
Based on Radiation Dose Modeling
and Multi-criteria Decision Analysis
for Oskarshamn Nuclear Reactor 3
Lessons Learned from Operating
Experience in the Reuse
of Decommissioned Sites

May AlAli
Abstract

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An increasing number of decommissioning activities are being undertaken worldwide at facilities where radioactive material has been produced, used, or managed, which will also be the case of the Oskarshamn nuclear reactor O3 in Sweden, the remaining operational unit at that power plant. Decommissioning, which seems to be the sign of ending, is, in fact, a future-oriented process that can prove successful the safe management of nuclear facilities throughout their entire lifetimes but especially after shutdown. In this context, this research aims to develop a comparative profile of decommissioning scenarios based on a radiation dose modeling approach using RESRAD-BUILD software. Conducting then a multi-criteria decision analysis (MCDA) of scenario comparison for the decommissioning of O3 is essential for updating ongoing decommissioning projects, making strategic decisions for future ones, and bringing in sustainable site reuse options. The exposure-to-dose model will be used to evaluate and optimize safety parameters, site release levels particularly, related to personnel and the environment. Testing multiple sets of parameters in the decommissioning plan will be used to compare results and to assess the sensitivity of the strategy to variable inputs. This comparison with the MCDA Analytic Hierarchy Process model (AHP) results will allow the identification of the most optimal reuse scenario for Oskarshamn unit 3 after its useful life ends. Also, the lessons learned from operating experience in the reuse of decommissioned nuclear sites can be incorporated systematically into the eventual decommissioning of O3, and which, according to current planning will run until 2035 or 2045.
An increasing number of decommissioning activities are being undertaken worldwide at facilities where radioactive material has been produced, used, or managed, which will also be the case of the Oskarshamn nuclear reactor O3 in Sweden, the remaining operational unit at that power plant. Decommissioning, which seems to be the sign of ending, is, in fact, a future-oriented process that can prove successful the safe management of nuclear facilities throughout their entire lifetimes but especially after shutdown. In this context, this study aims to optimize radionuclide site-specific concentration limits, called Derived Concentration Guideline Levels (DCGLs), and which are used to guide the cleanup of a nuclear decommissioning site to meet radiological criteria. Dose trends for eight radionuclides are calculated for four dose assessment scenarios: demolition, nuclear industrial, non-nuclear industrial, and residential. The considered radionuclides are Cs-137, Cs-134, Sr-90, Ni-63, Co-60, Fe-55, C-14, and H-3. Nuclide-specific clearance levels and total clearance levels for each scenario are calculated and compared with those obtained from four previous studies on clearance levels for the reuse of materials and buildings of a nuclear power plant. This comparison validates the results by showing that the calculated DCGLs for all of the radionuclides, except Co-60, are indeed less stringent than the generic risk-based ones established by the IAEA and in other studies, no matter whether considering industrial or residential use. And so, these DCGLs are optimized values as they give the real minimal remediation levels required and hence optimize the release process and the associated costs. Optimized safety parameters, site release levels particularly, with the Multi-criteria Decision Analysis (MCDA) results show that adaptive reuse is the next prime alternative to adopt for a decommissioned nuclear facility/site. Now whether industrial redevelopment for nuclear or non-nuclear use highly depends on the most recent trends in energy followed by cost and environmental impact. Sweden, our country of interest, is moving away from nuclear. It is withdrawing all its reactors from service by the early 2040s, aiming for a 100% renewable electricity supply by then. That is due to a lot of factors, including the fact that new nuclear power plants have proved to be expensive in recent years, often involving delays and cost overruns, and that their eventual decommissioning is viewed and dealt with as a terminal phase. Based on all the results, the most optimal use of the site at release is industrial non-nuclear. This study reiterates that to be able to integrate the different arguments and reach this conclusion, it is necessary to update the process of license termination and site release where the endpoint should go beyond site release and consider post-decommissioning reuse.
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1. Introduction

1.1 Background and Rationale

The International Atomic Energy Agency’s (IAEA’s) Power Reactor Information System (PRIS) database [1] reported—as of the beginning of 2020—441 operational nuclear power reactors in the world, with a total net capacity of 389,979 MWe, accounting for about 10.15% of the total gross electricity production and one-third of all low carbon electricity. 173 permanently shut down, 153 in decommissioning process or decommissioned, and 53 under construction (fig.1). Approximately 20,000 workers are employed in the nuclear power sector of a given national economy over the lifetime of a gigawatt of nuclear electrical generating capacity. [2]

![Figure 1 The most recent reactor data available in 2020. (PRIS, 2020)](image1)

![Figure 2 The life cycle of a nuclear power plant. (NEA, 2018)](image2)
In the context of a nuclear facility’s life cycle shown in fig. 2, planning for decommissioning – which appears to be the sign of ‘ending’ – is, in fact, a future-oriented activity characterized by the intended end-use application that is to be for the site, which may not involve the removal of all radioactive material. [3] Therefore, the next use may be restricted or unrestricted depending on the nature of the end state and how much material has been removed, but is, in any case, protective of both people and the environment.

Decommissioning of nuclear facilities (nuclear power plants, research reactors, fuel fabrication plants, etc.) is an extensive and multidisciplinary task that touches all the technical, financial, and social aspects of nuclear power and whose sphere of influence is at least as large as that of nuclear power. Decommissioning as specifically applied to NPPs goes far beyond techniques and apparatus because it involves the special problems of handling radioactivity and because power plants are often heavily linked to community economies. [3] The decommissioning process includes partly or fully dismantling of components and structures, decontamination and clearance, demolition of buildings in some cases, remediation of any contaminated ground, and finally a survey of the site to be released from the regulatory framework. [4] Several of these activities generate radioactive or potentially radioactive waste, which must be managed properly before clearance or disposal.

The varied nature of activities to undertake, their interfaces, and interdependencies affect the management and technical actions associated with transferring the facility from an operating plant to an object under decommissioning. [3] That’s why, many preparatory activities for decommissioning and dismantling, such as asset management, spent fuel and fissile material management, waste management, the closing of redundant systems, and decontamination activities should begin at the early stages of facility development i.e. the plant design phase and should continue through to the termination of the license. [5] (fig. 3 and 4)

To aid in consistent and efficient performance of these activities, a decommissioning plan is developed to explain how the facility will be safely dismantled, how radiation protection of workers and the public is ensured, how environmental impacts are addressed, how radioactive and non-radioactive materials are managed, and how the regulatory authorities for the facility and site are to be terminated. [5] Nuclear decommissioning is regulated by statutory provisions, incorporated in, or based on (1) the Act on Nuclear Activities, 1984, (2) the Radiation Protection Act 1988, and (3) the Environmental Code 1998. [6]

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1 The term ‘facility’ as used in this thesis means a facility with its associated land, buildings and equipment in which radioactive material is used, processed, handled or stored.
2 The Swedish Radiation Safety Authority (SSM) (6) defines dismantling of contaminated and/or activated equipment, components and structures as either irreversible (dismantling) or reversible (disassemble) measures for the purpose of decommissioning the facility.
3 Demolition is defined by SSM (6) as irreversible measures applied to contaminated and/or activated building structures or buildings for the purpose of decommissioning the facility.
Figure 3 A typical organizational chart for a nuclear power plant. (IAEA, 2019)

Figure 4 The different preparatory activities to transfer a nuclear facility from operation to decommissioning. (NEA, 2018)
The significant uncertainties in decommissioning, site remediation and the long timescales involved in most tasks have been widely demonstrated in the literature. [3] [4] [5] Success in dealing with these challenges requires, in addition to innovation and improvement through research and development, attention to decisions on sites’ use after service, something current decommissioning projects did not value highly. Next is an empirical analysis of what explains why the definition of a clear site redevelopment early on is important.

The reason why nuclear power plants have to be decommissioned is the induced radiation and henceforth the need to achieve a condition of no radiation hazard or adverse consequences for human health and safety, quality of life, property, and the environment. The regulatory body and other competent authorities will continue to supervise the facility that is to be decommissioned to ensure its fulfillment of protection and safety responsibilities to people and the environment, both during and after decommissioning. This, of course, entails strict specifications and overarching requirements on how the facility is to be decommissioned, and very likely, high costs and extra efforts throughout a whole ‘non-productive’ period from which it is unlikely possible to create value in a ‘conventional system’. That is to say, conventional decommissioning requires operators to divert cash and human resources away from revenue-generating activities and removes the option to reuse the site in the future. And if for some reason, and these could be several, there won’t be enough funds for immediate dismantling, the cost will be borne by the future generations or the decommissioning will be done at a less suitable time or not at all. This likely allows a greater margin for error and, hence, higher, unfairly distributed total costs due to inflation, real interest rates, and the follow-on cost of postponing the cleanup.

No one wants that.

That’s why key emerging research started on finding a way to achieve decommissioning goals more efficiently not only to ease this financial burden now but to create value in the future. How? It is strongly anticipated that evaluating redevelopment and reuse opportunities can be the way to more innovative decommissioning. Effective decommissioning – and, before that, advanced planning for the site’s future potential – are key to optimizing release criteria, maximizing cost savings, and turning decommissioning into a business opportunity. Selecting a specific redevelopment option will have to consider several factors of diverse character, but the place to begin, this thesis goes, is with radiological modeling to convert residual radioactivity contamination levels to radiation doses and, hence, help decide on what future use of the facility to implement in the specific context of health and safety. The other relevant factors are accounted for in Section 5.2 on MCDA.

Early dismantling to a green (unrestricted release with or without structures remaining) or brownfield (restricted release) without a safe enclosure period is usually the preferred method for NPP decommissioning, which, of course, has both advantages and disadvantages, but is adopted here because the priority is to take advantage of the knowledge that exists now. Naturally, the decontamination of the reactor, equipment,
and buildings channels to a safe, fast, and more cost-effective dismantling. For this purpose, an alternative to segmenting the reactor pressure vessel is to remove and transport it in one piece. This study assumes this method because:

- it reduces the cost of handling operations;
- it shortens the dismantling schedule by around 11 months which results in cost reduction, as certain time-dependent costs would no longer accrue;
- it reduces occupational radiation dose to dismantling/demolition workers; and
- barge shipment reduces public radiation dose in principle.

Directly after, the plant can be released for either restricted use, which could be nuclear or non-nuclear, or for unrestricted use, which could be the unconditional clearance of buildings for reuse or demolition. Operators follow the ‘conventional’ steps of the process flow diagram shown in fig. 5 where the endpoint is site release.

Over the past decade, a large number of studies [7] have documented the successful release of nuclear sites by using the two dose criteria specified in IAEA safety series No. 89 [8] for site exemption from regulatory control. The first criterion is defined as the de minimis level of individual dose, ranging from few tens of microsieverts to 0.1 millisieverts per year based on an ‘acceptable’ risk level (negligible compared with other risks). The second one is defined as the de minimis collective dose, set at a larger fraction of one person-sievert per year of practice and below which a formal optimization analysis is not needed. Here are two points worth heeding: (1) although these de minimis dose rates should be considered in deriving radionuclide concentration guidelines, few sites were concerned with that [9], and (2) achieving de minimis levels, however, does not automatically lead to exemption from regulation. The latter extends to broader social and economic factors that include, but are not limited to, the purely risk-based factors handled by the de minimis dose.

Despite this, notices of the nuclear safety and security commission [10] clearly state that the most important area for releasing a nuclear site is its release from the requirements for radiation protection of the appropriate regulatory body. Likewise, Larsson and Weber [11] assert that it is the basis for planning, identifying the extent and nature of the contamination, assessing potential risk impacts, estimating the cost, and controlling material arising from decommissioning, as well as supporting decisions after site release.

Concerning the first point in connection with the de minimis dose, several studies [9] [12] [13] [14] contributed several reasons for calculating and implementing site-specific Derived Concentration Guideline Levels (DCGLs). While generic release levels, used to guide cleanup, are based on generic pathway analyses and conservative estimates of parameters, DCGLs are more realistic, being determined by likely scenarios and site-specific parameters, without compromising health and safety requirements. [15] The excluded scenarios are not included in the projected site use until the radioactivity decays to a permissible level. This process proved to yield the most direct measure of success of the decontamination operation and can be customized to the site by using applicable variables and future-use scenarios. [15] This way, less decontamination is secured,
the amount of Final Status Surveys (FSS) sampling is reduced, and on-site analysis of FSS samples (using gamma spectroscopy) is possible. Here appear to be opportunities for major cost savings from lowered decontamination and remediation costs, rapid sampling, reduced costs associated with the analysis of the samples, and faster delivery of the FSS report. This is what researchers broadly mean by applying the “ALARA principle” and making the best use of available resources, which require a cost-benefit analysis or an equivalent assessment to determine if the individual or collective doses can be reduced and at a reasonable cost. Other than the benefits just mentioned, the derivation of site-specific release levels is important in countries where no generic ones are available, and for those radionuclides that do not have generic acceptance limits. That's why, and in the absence of a need for international harmonization of release criteria, different models for the derivation of site-specific release criteria have been developed and, in some cases, applied.

Hence now, when planning the decommissioning of an NPP, as well as the design, construction, and operation of a new one, the importance lies not only in the experience and lessons learned of decommissioning of existing power plants but also in planning and decision-support tools using computer simulation technique. Questions tackled include radiation dose minimization, balancing cost reduction and safety, and performing activities in nuclear installations, in and after service.

In the IAEA’s Nuclear Safety Review 2019 [17], the acting director-general confirmed that, despite the increasing flow of decommissioning projects, only a limited number has been recently released, either because they have not reached a state where the release of the site is imminent yet or because the site is, or will be, reused for nuclear activities. Here, the concept of ‘reuse’ is grounded, but shyly, as it is for nuclear purposes (again). This seems to violate what would be acceptable in multi-dimensional decision making for complex situations. Ideally, as highlighted by Flüeler [18] in his book on Decision Making for Complex Socio-Technical Systems, the nuclear option must be compared against other options and be selected only if optimal for the site. Yet, little published guidance on approaches to applying multiple reuse options to choose the most suitable one was found.

Also, not only is there a lack of published information on redevelopment/reuse options and their role in the nuclear energy sector, the complex process of planning one when the facility is already shut down compounds the problem. Although a book by Laraia, IAEA scientific secretary, published in 2019, [19] explains this role and the different involved factors, evidenced by various cases of successful decommissioning remediation projects resulting in the redevelopment of nuclear sites worldwide, its findings have been followed only selectively. Operators with upcoming decommissioning projects continue to overlook redevelopment opportunities and sustainable practices early in the project life cycle, which often leads to time and cost overruns.

While the IAEA published only two Technical Report Series on the redevelopment of nuclear facilities and sites after decommissioning, the most recent being in 2011, Laraia asserts that this topic “is something to which we at the IAEA attach considerable importance, and we intend to continue working in what we believe will be an increasingly significant area.” [20]
Figure 5 The ‘conventional’ steps of the process of license termination and site release under the immediate dismantling strategy for decommissioning of a nuclear power plant. (NRC, 2019)
As is clear from the above, much of the existing literature on decommissioning addresses its technological aspects and emphasizes the importance of meeting safety requirements, and little of the research dealt with reuse and redevelopment, if indeed any mention. It is this thesis aspiration to extend that work and use safety, an already established priority in nuclear, in planning for the reuse of nuclear facilities and their sites.

The rationale for a focus on reuse is premised from the generally assumed view of decommissioning as an end-of-life costly phase that requires constant and lengthy efforts to achieve, imposing as high-cost and difficult tasks related to the management of the radioactive waste on future generations. The focus on Sweden stems from the fact that no reported research in the area of the reuse of decommissioned sites incorporates any Swedish experience or perspectives. The method of the case study is perceptively applied to test theoretical predictions of post-decommissioning conditions and guide practical decision making. While this thesis may be of special interest to operators of Oskarshamn NPP to assist them in approaching the challenge to decommission the facility and restore the site for alternative reuse, its findings mean to offer tips to current and potential future nuclear operators who wish to understand and benefit from the advantageous experience of evaluating reuse options early on. This endeavor was given further credibility by the positive and supportive feedback from senior advisors at OKG, who were consulted in the early stages of the research regarding its value and appropriateness.

1.2 Research question
What is the most suitable reuse scenario for the third reactor O3 of the Oskarshamn nuclear power plant in Sweden after decommissioning?

1.3 Problem statement
The lack of timely and decisive strategic planning for the site’s end-use after decommissioning by nuclear operators.

1.4 Aim and Objectives
This thesis is conceived as a project under the Energy Studies Master’s Programme. It aims to identify and justify how the remaining operating unit of Oskarshamn NPP can be optimally reused in the post-operational phase. On a direct analytical level, this thesis aims to create a better understanding of the importance of deriving site-specific release levels and highlight the power of the reuse concept to break through the decommissioning “dead-end”.

The objectives of the thesis are:

- to bring information pertinent to priorities, activities, and methods to be considered in attaining the goal of safe and facilitated decommissioning;
- to simulate several exposure pathways and scenarios based on three design phases: scenario development, radiation mapping, and derivation of release levels; and
- to assist decision making for the selection of the optimal post-decommissioning option.
1.5 Scope
The scope of this thesis is to present the different possibilities introduced into
decommissioning planning by simulating and assessing exposure dose after the
decommissioning of the third unit of the nuclear power plant in Oskarshamn, Sweden under
the principle of ALARA per radiological environment change. The following aspects are
presented in the paper:

- RESRAD-BUILD computer code as a pathway analysis model for routine work and
  interventions in 3D decommissioning environments with risk of radiation exposure,
  and for application in decision making in potential post-decommissioning
  redevelopment/reuse
- Modeling, evaluating, and optimizing decommissioning scenarios as for the
  selection of the optimal option using the Analytic Hierarchy Process (AHP) in
  Multicriteria Decision Analysis (MCDA)
- Evaluating radiological safety in decommissioning

1.6 Structure
This thesis is organized into seven sections. Section 1 (this section) contains background
information based on a literature review and defines the aim, objectives, and scope of the
study. Section 2 explains the applied methodology for modeling and evaluating
decommissioning scenarios using computer code in support of radiological protection in the
field of nuclear decommissioning. It then introduces the AHP as an MCDA method
for ordering alternatives. Section 3 provides a general site description of the Oskarshamn
Nuclear Power Plant (NPP). Section 4 describes the four post-decommissioning scenarios
considered for Oskarshamn unit 3 and the input parameters for each, based on acquired data
from different inventories and judgments of experts in the field. Section 5 presents the
results of applying the methodology to the site. Section 6 is the discussion of results
to compare the scenarios and select the most optimal one. Section 7 summarizes the
conclusions of the study and suggests directions for future work.

2. Methodology
The background information presented in this study is based on a review of international
publications. The current study also implements secondary analysis of existing data from
previous decommissioning studies of Oskarshamn NPP to be used as the input data for
applications with the computer program RESRAD-BUILD, which assists in producing and
comparing exposure scenarios based on dose assessment and dose optimization. At the end
of the study, an analytic hierarchy process model is developed to evaluate the resulting
decommissioning scenarios based on both quantitative and qualitative considerations and
multi-criteria analysis. Two site visits to Oskarshamn NPP are intended to supplement the
data used in the study through observation and semi-structured interviews.

4 The Analytic Hierarchy Process (AHP) is a theory of measurement through pairwise comparisons that relies
on the judgments of experts to derive priority scales.
2.1. RESRAD-BUILD computer code

2.1.1. Overview and Justification
ALARA calculations and dose assessment for work planning in complex nuclear facilities are difficult, especially in decommissioning sites where the geometry, source distribution, and shielding configuration of the working area are constantly changing. The optimization of radiological protection in the nuclear industry and the area of decommissioning is an important part of the safety culture, which is why several work scenarios need to be considered and compared before deciding on the final one. In order to create, edit, and select between various alternative scenarios for a specific use, based on 3D-geometrical, material, and radiological information, this study will use the RESRAD-BUILD computer code. RESRAD-BUILD is an exposure pathway analysis model that belongs to the RESRAD family codes developed by Argonne National Laboratory (Argonne) under the U.S. Department of Energy. [21] While a number of computer codes are available for dose modeling, the RESRAD code with its excellent implementation, description, and quality assurance has become a quasi-standard for site release over the past years in many countries, and many nuclear site licensees and operators of decommissioning projects check their approaches against RESRAD results. [22] Since its original release, this code has undergone 14 major updates, the latest of which – RESRAD-BUILD 3.5 is used in this work. This particular software is adopted because it is one of the most practical and user-friendly software to assess the radiological dose incurred by hypothetical receptors in radioactively contaminated sites and to calculate site-specific Derived Concentration Guideline Level (DCGL) values from activity/dose relationships at decommissioning sites through various exposure pathway scenarios, decay-time intervals, and radionuclides. [14] DCGL is defined by the U.S. Environmental Protection Agency (EPA) [23] as a radionuclide-specific activity concentration within a survey unit corresponding to the ‘release criterion’ that is a regulatory limit expressed in terms of dose or risk.

2.1.2. Methodology
The RESRAD-BUILD computer code employs two main models: the indoor air quality model for radioactive material release mechanisms and transport in the indoor environment and the external radiation exposure model for exposure assessment and scenario building in the 3D display. [21] It accounts for site-to-site variation in buildings — the different structural materials, sizes, exchange rates within buildings and their compartments, and the differences in size, shape, and thickness of contaminants inside the buildings. [21] It also incorporates both deterministic and probabilistic/uncertainty dose analyses, computing as such the total uncertainty induced in the output (dose) due to either the uncertainty in- or the probabilistic nature of the input parameters in decommissioning. [21] The code illustrates a building system with up to 3 compartments, 4 source geometries (point, line, area, and volume), 10 source locations and10 receptor locations in a single run. [21] The volume source can have up to 5 layers, each being homogeneous and isotropic. [21] A certain amount of specific shielding material can be inserted between each set of source and receptor (s) for external gamma dose calculations. [21] The results can be shown in both text and graphic reports.
2.1.2.1. Exposure pathways

The RESRAD-BUILD code assesses exposure to radionuclides emitted to the indoor air through seven plausible exposure pathways (fig. 6):

1) external exposure directly from the source
2) external exposure to contamination on the ground, i.e. air submersion
3) external exposure due to submersion in a contaminated atmospheric cloud, i.e., air submersion
4) inhalation of airborne radioactive particulates
5) inhalation of aerosol indoor radon progeny and tritiated water vapor
6) unintentional ingestion of radioactive material directly from the source
7) unintentional ingestion of deposits on the surfaces of the building compartments

External exposure is characterized by the first three pathways, whereas the last four relate to internal exposure due to internal contamination. The code generates the external exposure as the Effective Dose Equivalent (EDE)\(^5\) and the internal exposure as the Committed Effective Dose Equivalent (CEDE)\(^6\), the sum of both of which is the total radiation dose, expressed as the Total Effective Dose Equivalent (TEDE).

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\(^5\) The U.S. Nuclear Regulatory Commission defines the EDE as the sum of the products of the dose equivalent to the organ or tissue (\(H_T\)) and the weighting factors (\(W_T\)) applicable to each of the body organs or tissues that are irradiated (\(HE = \Sigma W_T H_T\)).

\(^6\) The U.S. NRC defines the CEDE (\(H_{E,50}\)) as the sum of the products of the committed dose equivalents for each of the body organs or tissues that are irradiated multiplied by the weighting factors (\(W_T\)) applicable to each of those organs or tissues (\(H_{E,50} = \Sigma W_T H_{T,50}\)).
radiological analysis of contaminated buildings includes also wound exposure and absorbed skin dose under internal exposure. However, and because their contribution to the radiation dose is much smaller than that of the above-mentioned pathways [25], RESRAD-BUILD does not take account of these two pathways but does estimate the dermal absorption of tritium by increasing its inhalation dose conversion factor by 50%. [21]

2.1.2.2. Exposure scenarios

The release of buildings and sites is one of the last steps in the preparation of a decommissioning programme for NPPs. Any decommissioning option, from placing the plant in a stand-by condition, to the ultimate removal of the facility and restoration of the site, must meet the requirements of protecting the environment and the public’s health and safety according to site release criteria. The RESRAD-BUILD code is used to derive suitable release criteria that are known to be directly influenced by radiological dose criteria. [16] It then simulates a set of possible future uses, defined as exposure scenarios, and if the site has been cleaned up sufficiently for the release criteria, the site should be released for unrestricted use, which is the preferred option. RESRAD-BUILD allows for the adequate selection of scenarios for the future use of the site after release from controls. These scenarios differ along three drivers: (1) the amounts of contaminants and the rates at which they are emitted into the air, (2) the exposure time, and (3) the exposure pathways implicated. The user constructs building occupancy and building renovation scenarios by influencing the entry of six major input parameters: Case, Building, and Source parameters with options to access more detailed input windows, Shielding parameters with options to specify multiple shields and shielding properties, Receptor parameters, and Radiological Units. [21] For any scenario, dose conversion factors (DCFs) for external inhalation and ingestion are taken from the U.S. Environmental Protection Agency Federal Guidance Report No. 11 (FGR-11)[7], and for direct external exposure and air submersion from FGR-12[8]; radionuclide half-lives are taken from the International Commission on Radiological Protection Publication 38[9]. [21]

2.1.2.3. Input parameters

2.1.2.3.1. Case window

The case window displays two of three exposure time parameters: the Exposure Duration, that is the period over which the exposure happens, and the Indoor Fraction, that is the time fraction [of the exposure duration] during which a receptor is inside the building. [21] The Receptor parameters include the third time parameter, the Time Fraction. This allows

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entry of receptor-specific time fractions, the product of which is the exposure time of a receptor at a given location. [21] The calculated radiological dose is the total dose received during this exposure time. The code permits the calculation of the time-integrated dose, as well as the dose rate, hence providing the doses for the user-specified scenario at every time step i.e. at the initial characterization and at later times. [21] The user can also specify how many times (up to 10) to be evaluated in the Evaluation Times window. (21) The value for the initial time t₀ is automatically entered. The user selects either 1, 2, 3, 5, 9, 17, 33, 65, 129, or 257 as the number of receptor points inside the building to calculate the time-integrated dose; selecting 1 generates the instantaneous dose at the user-specified time. [21]

2.1.2.3.2. Building parameters
The conceptual building model takes in the characteristics of the building and produces one-to-three-room models, wherein the area and the ceiling height of each room must be input. As will be brought out later in this chapter, these parameters are used in the air quality model, which, to be solved, requires the entry of the average building air exchange rate for the one-room model, and add to that, the outdoor inflow rates for the rooms and the flow between adjacent rooms for the two- and three-room models. [21] The solution to the deposition model depends on both the deposition velocity and the resuspension rate, assumed to be the same in each room. [21]

2.1.2.3.3. Receptor parameters
The Receptor Parameters window takes in the characteristics of an individual. It shows input for the individual specified by the receptor number and for both the ingestion and breathing rates. Up to 10 receptor points can be listed, each being characterized by the room, the location, and the time fraction spent at each location. Multiple receptors could be considered either as many different individuals in a room or as one individual being in multiple rooms. All receptors and sources require but a single reference origin (defined by the user) and according to which the receptor’s location is defined as the absolute coordinate of the receptor’s midpoint location. [21] The receptor’s location should be 1 meter above the absolute location where the receptor is standing for example. [21] This information is only of particular importance for the direct external exposure pathway. In most situations, however, the receptor is not standing still in front of the source but moving around; that’s why the code uses external DCFs for rotational orientation from FGR-12. [21]

2.1.2.3.4. Shielding parameters
The Shielding Parameters window takes in the characteristics of the shielding that goes between each source-receptor pair when calculating the external dose. The user can select the shielding material from 8 material types (concrete, water, aluminum, iron, copper, tungsten, lead, and uranium) and input its thickness and density. [21] The Shielding Parameters window shows the shielding parameters for the source and the receptor(s) that are being viewed in their respective windows, but which can also be copied to other sources and receptors.
2.1.2.3.5. Source parameters

The Source Parameters window takes in information about the contamination source type and location. The source location is determined by the room number and the coordinates of the source’s center point, which is the absolute coordinate according to the reference origin previously defined in the Receptor Parameters window. [21] Subwindows collect data on geometry (point, line, area, or volume), radon release (if relevant), size, removable fraction, air release fraction, and radionuclide contamination of the source. [21] The prime cause of having two direct exposure models for external dose calculations is due to having these different geometric types of sources. The one for the point and line sources, which is a simple dose integral method, and the one for area and volume sources, which is based on the semi-infinite slab approach that approximates area and volume sources to slabs. [21] This aspect will be dealt with in more detail in the subsection on External Radiation Exposure. Volume sources can be readily modeled as distinct regions by entering for every region the following properties: density, porosity, thickness, radon emanation fraction, radon diffusion coefficient, and the erosion rate. [21] It is important to note that for the tritium volume source, a separate source window has to be created, with tritium being the only contaminant.

2.1.2.3.6. Radiological Units

The Radiological Units window allows the user to choose the unit for the activity concentration: either curie (Ci), disintegrations per minute (dpm), disintegrations per second (dps), or the SI unit becquerel (Bq) and to view the resulting dose equivalent either in rem\(^{10}\) or in the SI unit sievert (Sv). [21]

2.1.2.4. Indoor air quality model

Radioactive materials in building structures are liberated into the indoor air by processes such as diffusion (radon gas and tritiated water), mechanical removal (decontamination activities), and/or erosion (removable surface contamination). [21] The indoor air quality model depicts the transport of these radioactive dust particulates, and radon progeny via (1) their deposition and resuspension, (2) air exchange between compartments and with outdoor air, and (3) radioactive decay and ingrowth. [21]

Its application in air exchange- and air infiltration calculations is to determine the concentration of specific radionuclides indoors and, hence, the total occupant exposure.

2.1.2.4.1. Building ventilation and infiltration

The ventilation (or air change/exchange) rate of a building (or a compartment) is the ratio of the volumetric rate at which air enters (or leaves) the building (or the compartment) divided by the volume of the building (or the compartment). [26] It is usually expressed in

---

\(^{10}\) The U.S. NRC defines rem (Roentgen equivalent man) as one of the two standard units used to measure the dose equivalent (or effective dose), which combines the amount of energy (from any type of ionizing radiation that is deposited in human tissue), along with the medical effects of the given type of radiation. the dose equivalent (in rems) is equal to the absorbed dose (in rads) multiplied by the quality factor of the type of radiation [see Title 10, Section 20.1004, of the Code of Federal Regulations (\textit{10 CFR 20.1004}), "Units of Radiation Dose"].
air changes per hour (ACH) (SI unit is h\(^{-1}\)). A building with a ventilation rate of 1 h\(^{-1}\) for instance has its volume of air replaced once every hour. Ventilation is either natural (airflow through open windows, doors, and other designed openings in the building) or mechanical (controlled air movement driven by fans).

Note that RESRAD-BUILD represents a building by three compartments \((N = 3)\) maximum and assumes that compartment 1 is not adjacent to compartment 3, and, thus, air exchange between compartments 1 and 2 and 2 and 3 is possible, but not between compartments 1 and 3. All compartments can exchange air with the outdoors, however.

From the steady-state compartment mass balance equation:

\[
\sum_{j=0}^{3} Q_{ji} = \sum_{j=0}^{3} Q_{ij} \quad (1)
\]

Where \(i\) or \(j\) = compartment index, representing the outdoor air and compartments 1, 2, 3 for \(i = 0, 1, 2, 3\) respectively (compartment 0 is the outdoor air)

\(Q_{ji}\) = flow from compartment \(i\) to compartment \(j\) (m\(^3\)/h)

\(Q_{ij}\) = flow from compartment \(j\) to compartment \(i\) (m\(^3\)/h)

The air exchange rate for compartment \(i\) is calculated as:

\[
\lambda_i^a = \frac{\sum_{j=0}^{3} Q_{ji}}{Vol_i} = \frac{\sum_{j=0}^{3} Q_{ij}}{Vol_i} \quad (h^{-1}) \quad (2)
\]

Where \(\lambda_i^a\) = air exchange rate (h\(^{-1}\))

\(Vol_i\) = volume of the building enclosure (m\(^3\))

The building air exchange rate is:

\[
\lambda_b^a = \frac{\sum_{i=1}^{3} Q_{i0}}{\sum_{i=1}^{3} Vol_i} \frac{\sum_{i=1}^{3} Q_{0i}}{\sum_{i=1}^{3} Vol_i} \quad (h^{-1}) \quad (3)
\]

Infiltration rate is the volumetric flow rate of outside air into a building through cracks, interstices, and other unintentional openings in any of its elements, typically in cubic meter per minute (CFM) or liters per second (LPS). (26) It can also be expressed in ACH.

It is calculated based on the results of statistical fits to long-term time-series data of infiltration rate measurements and associated climatic data. (26) In its most basic form, the air infiltration rate is expressed as a linear function of wind and temperature:

\[
Q_{inf} = a' + b' \Delta T + c' v^2 \quad (h^{-1}) \quad (4)
\]

Where \(Q_{inf}\) = infiltration rate (h\(^{-1}\))
\[ \Delta T = \text{internal/external temperature difference (°C)} \]

\[ v = \text{wind speed (m/s)} \]

\[ a', b', c' = \text{regression coefficients} \]

Known combinations of \( \Delta T \) and \( v \) are substituted in the equation, and regression coefficients are calculated by the method of ‘least squares’\(^{11}\) and are unique to each building.

### 2.1.2.4.2. Airborne concentrations of radionuclides

In general, ventilation rate and effectiveness differ from one contaminant to another due to contaminant-related factors primarily: [26]

- source and strength of the contaminant
- external concentrations
- discomfort effects
- toxicity

As a rule, the minimum prescribed ventilation rate must be able to meet the rate required to disperse the contaminant that needs most ventilation. In the case of multiple-source contaminant releases, the ventilation rates needed to fulfill the requirement of each source are summed to obtain the total rate. [26] The model assumes the condition of uniform mixing of particulates in the indoor air within each compartment of the building, and, hence, the contaminant concentration is the same at every point in the air within the compartment. [21] Therefore, the model is capable of estimating the time-varying concentration of these contaminants.

Assuming that the contaminant in the indoor air is well mixed in the building (or compartment) enclosure, the concentration of a radionuclide is given at any point in time by the continuity equation:

\[
Vol \times \frac{dC_{in}}{dt} + mQ(C_{ext} - C_{in}) = (S - \lambda_{rn}) \text{ (mass/time) (5)}
\]

Where \( Vol = \text{volume of the enclosure (m}^3) \)

\[ m = \text{empirical mixing factor (varies from 0 to 1, } m = 1 \text{ is equivalent to perfect mixing)} \]

\[ Q = \text{ventilation rate (m}^3/\text{s) } \]

\[ C_{in} = \text{internal concentration of contaminant (mass/m}^3) \]

\[ C_{ext} = \text{external concentration of contaminant (mass/m}^3) \]

\(^{11}\) The method of Least Squares is a procedure to determine the best fit line to data.
\( S = \) total rate of contaminant release (from all sources)

\( \lambda_{rn} = \) rate of radioactive decay of radionuclide \( n \) (mass/s)

Assuming a steady-state air exchange rate, the internal concentration of contaminants will eventually reach equilibrium. Thus \( \frac{dC_{in}}{dt} \) will tend to zero and the equilibrium concentration will be given by:

\[
(C_{ext} - C_{in}) = \frac{(S - \lambda)}{mQ} \quad (6)
\]

In the case of radon, there won’t be an external component \( (C_{ext} = 0) \), and if perfect mixing is assumed \( (m = 1) \), the internal equilibrium concentration becomes:

\[
C_{in} = \frac{(S - \lambda)}{Q} \quad (7)
\]

Where \( \lambda = \) decay rate (of radon)

Once the emission of radionuclides into the indoor air ceases, the emission rate \( S \) becomes zero and the radionuclide concentration decays as a function of time. Eliminating \( (S - \lambda_{rn}) \) from equation and integrating yields:

\[
C_t = C_{t_0} e^{-\frac{Q}{v t}} \quad (8)
\]

Where \( C_t = \) contaminant concentration at time \( t \) after cessation of radionuclide emission \( t_0 \)

\( C_{t_0} = \) equilibrium radionuclide concentration

Equation (5) can be written as the equation of the mass balance of the contaminant within a volume \( V_i \) due to the contribution of a specific source:

\[
V_i \frac{dC_i^n(t)}{dt} = \lambda_{rn} V_i C_i^{n-1}(t) + \sum_{j=0, j \neq i}^{3} (Q_{ji} C_j^n(t)) - C_i^n(t) \sum_{j=0, j \neq i}^{3} Q_{ij} + I_i^n(t) - S_i^n(t) \quad (9)
\]

Where \( i = \) compartment index \((i = 0, 1, 2, \) and 3\) representing respectively the outdoor air (0\textsuperscript{th} compartment) and the first, second, and third compartments

\( C_i^n = \) concentration at time \( t \) of a radionuclide of order \( n \) in its radioactive decay series, present in the air of compartment \( i \) (pCi/m\(^3\))

\( \lambda_{rn} = \) radioactive decay constant of radionuclide \( n \)

\( Q_{ij} = \) flow of air from compartment \( i \) to compartment \( j \) \((i \neq j)\) (m\(^3\)/h)

\( I_i^n(t) = \) injection rate of radionuclide \( n \) into the air of compartment \( i \) at time \( t \) (pCi/h)
\[ V_i = \text{volume of compartment } i \ (i = 1, 2 \text{ and } 3) \ (m^3) \]

\[ S^n_i(t) = \text{sink term for radionuclide } n \text{ in compartment } i \text{ at time } t \ (\text{pCi/h}) \]

Each term of equation (9) represents the variation in time of a portion of the mass of the material within the volume \( V_i \) due to some process. The term on the left side of the equation denotes the net change (accumulation or depletion) of the pollutant within a volume \( V_i \) per unit of time. The first term on the right side of the equation denotes the formation of the principal radionuclide of order \( n \) due to the radioactive decay of its predecessor of order \((n - 1)\). The second term on the right side denotes the total contribution due to the inflow of air from the neighboring compartments. The third term denotes the loss of material due to the total outflow of air into the neighboring compartments. The sink term \( S^n_i(t) \) denotes the loss of mass of airborne material due to radioactive decay, \( S^n_{ri} \) and to material deposition, \( S^n_{di} \), and can be expressed as:

\[ S^n_i(t) = S^n_{ri}(t) + S^n_{di}(t) \ (10) \]

Where \( S^n_{ri}(t) = \lambda_{rn} V_i C^n_i(t) \) (11)

\[ S^n_{di}(t) = \lambda_{di} V_i C^n_i(t) \] where \( \lambda_{di} = \left( \frac{u_d A_i}{V_i} \right) \) (13)

Where \( \lambda_{rn} = \text{radioactive decay constant of radionuclide } n \ (h^{-1}) \)

\( \lambda_{di} = \text{deposition rate (h}^{-1}) \)

\( u_d = \text{deposition velocity (m/h)} \)

\( A_i = \text{horizontal area of compartment } i \ (m^2) \)

Therefore, the sink term \( S^n_i \) can be expressed as:

\[ S^n_i(t) = (\lambda_{rn} + \lambda_{di}) V_i C^n_i(t) \ (14) \]

The term \( I^n_i(t) \) in equation (9) represents the total injection of material (radionuclide \( n \)) into the air of compartment \( i \) due to the source present within the compartment, \( I^n_{Si} \), and due to resuspension of the material deposited on the floor, \( I^n_{hi} \). It can be expressed as:

\[ I^n_i(t) = I^n_{Si}(t) + I^n_{hi}(t) \ (15) \]

Where the source term \( I^n_{Si}(t) \) represents the direct injection rate of radionuclides into the indoor air. It depends on the mechanical removal or erosion rate of the source (or on the radon/tritiated water (HTO) diffusion process where there is a radon source or tritium volume source) and must be evaluated specifically for each defined source and operational scenario. (27) The resuspension term, \( I^n_{hi}(t) \) depends on the resuspension rate, \( \lambda_R \), and the deposition velocity, \( u_d \). [28]
2.1.2.5. **External radiation exposure**

The external radiation exposure model is designed as one function of the RESRAD-BUILD code to calculate the effective doses from (1) external exposure directly to the source, (2) external exposure to material deposited onto the ground (modeled as an area source), and (3) external exposure to aerosols and radioactive gases suspended in the indoor air (air submersion). [21] It is based on the point kernel method and the conversion coefficients for the surface (infinite plane) source, the infinite depth volume source, and the air submersion exposure given in the U.S. Environmental Protection Agency’s FGR-12 (refer to Appendix A). [29] These coefficients were derived by assuming that the contaminated media are large enough to be regarded as “infinite volumes” as compared to the range of the emitted radiations, which further implies that the energy emitted per cc of media is equivalent to the energy absorbed per cc of media. [30] All that is required is to convert MeV per disintegration to rem and to correct for the differences in energy absorption between air (or water) and tissue, and physical geometry of each specific exposure situation. [30]

2.1.2.5.1. **External dose from a contaminated volume source**

The total external dose from exposure to a volume source $V$ containing a radionuclide $n$ in compartment $i$ at time $t$ over the exposure duration $ED$, $D_{iV}^n(t)$, is expressed as:

$$D_{iV}^n(t) = (ED / 365) F_{in} F_{i} \overline{C}_{sv}^n(t) DCF_v^n F_G^n$$  \hspace{1cm} (16)

Where $ED = \text{exposure duration (d)}$

$365 = \text{time conversion factor (d/yr)}$

$F_{in} = \text{fraction of time spent indoors}$

$F_{i} = \text{fraction of time spent in compartment } i$

$DCF_v^n = \text{FGR-12 dose conversion factor for infinite volume source (mrem/yr per pCi/g)}$

$F_G^n = \text{geometrical factor for the finite area, source thickness, shielding, source material, and position of receptor relative to the source for radionuclide } n$

$\overline{C}_{sv}^n(t) = \text{average volume source concentration (pCi/g) of radionuclide } n \text{ over the exposure duration } ED \text{ starting at time } t$

It should be noted that, concerning the area source, it is treated as a volume source of small thickness (0.01 cm) with unit density. [21]

2.1.2.5.2. **External dose from a contaminated point source**

The total external dose from exposure to a point source $P$ containing a radionuclide $n$ in compartment $i$ at time $t$ over the exposure duration $ED$, $D_{iP}^n(t)$, is expressed as:

$$D_{iP}^n(t) = 24 ED F_{in} F_{i} \overline{C}_{sp}^n(t) \sum_j y_{nj} E_{nj} B(\mu \alpha t_a) d \left[ \frac{\mu_{en}(E_{nj})}{\rho} \right]_{air} \frac{e^{-\mu \alpha t_a}}{4\pi t_a^2}$$  \hspace{1cm} (17)

Where $24 = \text{time conversion factor (h/d)}$
\( ED = \) exposure duration (d)

\( \overline{C}_{ep}^n(t) = \) average total activity of radionuclide \( n \) in the source over the exposure duration starting at time \( t \) (pCi)

\( y_{nj} = \) yield for gamma \( j \) from radionuclide \( n \)

\( E_{nj} = \) energy for gamma \( j \) from radionuclide \( n \) (MeV)

\( d = \) unit dose rate per energy absorption

\[
\frac{\mu_{en}(E_{nj})}{\rho}_{air} = \text{mass energy absorption coefficient in the air (cm}^2/\text{g)}
\]

Where \( B \) and \( \mu \) are the buildup factor and the attenuation factor for the appropriate material (\( a \) for air, \( c \) for shield material, and \( s \) for source material) and which will be discussed at greater length in Subsection 3.2.2.

\[
B(x) = B_a \left( \frac{t_a}{t_a + t_c + t} \right) B_c \left( \frac{t_c}{t_a + t_c + t} \right) B_s \left( \frac{t}{t_a + t_c + t} \right) \tag{18}
\]

Where \( t \) is the source thickness, \( t_a \) is the air thickness, and \( t_c \) is the shielding thickness

2.1.2.5.3. **External dose from a contaminated line source**

The model for line sources differs from that of the point source in the more complicated geometrical factor.

The total external dose from exposure to a point source \( L \) containing a radionuclide \( n \) in compartment \( i \) at time \( t \) over the exposure duration \( ED \), \( D_{IL}^n(t) \), is expressed as:

\[
D_{IL}^n(t) = 24 \cdot ED \cdot F_{in} \cdot F_i \cdot \overline{C}_{sl}^n(t) \sum_j y_{nj} E_{nj} B(z') d \left[ \frac{\mu_{en}(E_{nj})}{\rho} \right]_{air} A_L \tag{19}
\]

\[
z' = \mu_a t_a' \tag{21}
\]

\[
A_L = e^{-\mu_c t_c} \int_{line} e^{-\mu_a t_a \sqrt{x^2 + t_a^2}} \frac{1}{4\pi (x^2 + t_a^2)} dx \tag{22}
\]

Where \( \overline{C}_{sl}^n(t) = \) average line source concentration of radionuclide \( n \) over the exposure duration \( ED \) starting at time \( t \) (pCi/m)

\( t_a = \) perpendicular distance to receptor

\( t_a' = \) distance from the receptor to the midpoint of the line source

These geometrical parameters are shown in fig.7 and are calculated based on the input geometrical parameters of the source and the location of the receptor.
2.1.2.5.4. External dose from contaminated dust in indoor air

The total air submersion external dose from exposure to indoor contaminated dust $P$ containing a radionuclide $n$ in compartment $i$ at time $t$ over the exposure duration $ED$, $D_{i, sub}^n(t)$, is expressed as:

$$D_{i, sub}^n(t) = \left(\frac{ED}{365}\right) F_{ini} F_i \overline{C_i^n(t)} DCF_{sub}^n$$ \hspace{1cm} (20)

Where $ED = $ exposure duration (d)

365 = time conversion factor (d/yr)

$\overline{C_i^n(t)} =$ average concentration of radionuclide $n$ in compartment $i$ at time $t$ over the exposure duration $ED$ (mrem)

$DCF_{sub}^n = $ air submersion dose conversion factor for radionuclide $n$ (mrem/yr per pCi/m$^3$)

The $DCF_{sub}^n$ is for a semi-infinite cloud source; no correction for the finite indoor air volume is conducted within the RESRAD-BUILD code. [21]

2.1.3. Photon attenuation coefficients

2.1.3.1 Linear attenuation coefficient

Consider the following situation. A beam of photons is incident normally on a thin uniform slab of material of thickness $dx$. Individual photons may pass through the slab without interacting, or they may be absorbed or scattered. The probability that an individual photon will interact in this thin section is given by the so-called linear attenuation coefficient $\mu$: 

(31)
\[
\mu = N_a \sigma = \frac{1000 N_A \rho}{A_r} \sigma \text{ (m}^{-1}) (21)
\]

Where \( N_a \) = the number of interaction-centers (atoms) per unit volume

\( \sigma \) = the total interaction cross-section per atom

For scattering by atoms, \( N_a \) may be calculated from the Avogadro constant \( N_A \), the atomic weight \( A_r \), and the density \( \rho \).

2.1.3.2 Exponential attenuation coefficient

A radiation field at a point P can be quantified by the physical nonstochastic quantity fluence, given by [32]

\[
\phi = \frac{dN}{da} \text{ (m}^{-2}) (22)
\]

where \( dN \) is the differential of the expectation value of the number of particles (whether photons or massive particles) striking an infinitesimal sphere with a great-circle area \( da \) around point P.

**Narrow beam attenuation.** Consider a material slab of thickness \( x \) and a narrow-collimated beam of photons passing through the matter. Part of the incoming photons will be removed from that beam, in a distance \( dx \), due to interaction processes like the photoelectric effect, Compton scattering, and pair production. [32] \( \Phi(x) \) is the fluence of photons that have not interacted in the slab after passing through the thickness \( x \).

The expected change \( d\Phi \) in the fluence after crossing a larger thickness \( dx \) is given by [32]

\[
d\Phi = -\Phi \cdot \mu \cdot dx (23)
\]

The linear attenuation coefficient \( \mu \) depends on the incident photon energy and the composition of the material and is influenced by the interaction processes previously stated.

Integrating:

\[
\Phi = \Phi_0 \cdot e^{-\mu x} (24)
\]

where \( \Phi_0 \) is the initial fluence. This equation, known as Beer’s law, describes the exponential attenuation of a photon beam, namely the uncollided photons that arrive at the dose point after traveling a distance through the material, also known as primary photons. [32]

**Broad beam attenuation.** The narrow beam attenuation determines only the number of uncollided photons arriving at the dose point; however, photons that were scattered in the medium or subsidiary photons locally emitted can also arrive at the dose point, and their contributions must be accounted for. [32] And so, a dimensionless correction factor \( B \), called the build-up factor, is introduced. The photon fluence after passing a distance \( x \) in a homogeneous medium can thus be written as: [32]
\[ \Phi = B(E, x). \Phi_0. e^{-\mu x} \quad (25) \]

The buildup factor \( B(E, x) \) is defined as the ratio of the total to the unscattered dose. [32] It is a function of the energy of the gamma radiation, and the distance traveled through the absorbing material, being proportional to the latter. [32] Buildup factors are essential in designing shields for radioactive sources such as power reactor cores. They are greater than one in value and approach to unity when absorption prevails or when the scattering cross-section disappears. [32]

**Photon fluence rate calculation.**

**Point source.** The photon fluence rate from a point source with activity \( A \), assuming one photon per disintegration, is given by [32]

\[ \Phi = \frac{S.B. e^{-b}}{4\pi \rho^2} \quad (26) \]

where \( S = \) the number of photons emitted by the source per unit time (n/s)

\( b = \) the main free paths. It is a dimensionless term that represents the attenuation effectiveness of a shield. The value of \( b \) is given by [32]

\[ b = \sum \mu_i \tau_i \quad (27) \]

Where \( \mu_i = \) the attenuation coefficient of material \( i \)

\( \tau_i = \) the distance traveled following the source-dose point line-of-sight through the material \( i \)

Note that the higher \( b \) is, the higher the radiation attenuation.

**Volume Source.** The photon fluence rate at a dose point near a volume source is determined by assuming that the volume source consists of several point sources. It is calculated by adding the contribution of every point source or “kernel” to the dose. The integration of this addition process is called “point kernel” integration: [32]

\[ \Phi = \int S.B. e^{-b} \quad (28) \]

### 2.1.3.3 Mass energy transfer- and mass-energy absorption coefficients

For dosimetric purposes, it is necessary to measure the energy transferred to secondary electrons as the result of the first interaction: [32]

\[ d(\hbar v)_{tr} = \Phi. h. v. \mu_{tr} \ dx \quad (29) \]
Energy $d(hv)_{tr}$ is transferred by interactions to kinetic energy (of electrons) when photons, of energy $hv$, travel a distance $dx$ in the material:

$$\mu_{tr} = \mu \frac{\langle T \rangle}{hv} \quad (30)$$

Where $\mu_{tr} = \text{the linear transfer coefficient}$

$\mu = \text{the linear attenuation coefficient}$

$\langle T \rangle = \text{the expectation value of the energy converted to secondary electrons}$

$hv = \text{the energy of a photon}$

Kerma ($K$) is energy transferred from uncharged particles to the matter: [32]

$$K = \frac{d(hv)_{tr}}{dm} = \frac{\Phi hv\mu_{tr}dx}{dm} = \Phi hv\left(\frac{\mu_{tr}}{\rho}\right) \quad (31)$$

Some of the energy transferred to the secondary charged particles is lost to radiative processes in the material, mainly Bremsstrahlung accounted to by the mass-energy absorption coefficient: [32]

$$\left(\frac{\mu_{en}}{\rho}\right) = \left(\frac{\mu_{tr}}{\rho}\right)(1 - g) \quad (32)$$

where $g$ gives the energy fraction lost to radiative processes.

### 2.1.3.4 Mass attenuation coefficient

The linear attenuation coefficient $\mu$ (m$^{-1}$) is dependent on density, which in turn is dependent on the physical state of the material. Consequently, $\mu$ is not the right quantity for data compilations, but the mass attenuation coefficient $\frac{\mu}{\rho}$ (m$^2$/Kg) that is independent of density. [32]

The four different mechanisms by which photons may interact with matter compete according to the probability of each. The total mass attenuation coefficient is, therefore, the sum of all the individual mass attenuation coefficients: [32]

$$\left(\frac{\mu}{\rho}\right) = \left(\frac{\tau}{\rho}\right) + \left(\frac{\mu_{coh}}{\rho}\right) + \left(\frac{\mu_{inc}}{\rho}\right) + \left(\frac{k}{\rho}\right) = (\tau + \sigma_{coh} + \sigma_{inc} + k)1000 \frac{N_A}{A_r} \quad (33)$$

The size of each attenuation coefficient will depend on the photon energy and the atomic number of the material.

### 2.2 Multi-criteria decision making: The Analytic Hierarchy Process

#### 2.2.3 Overview and Justification

Substantial research effort has been devoted to understanding and assisting the art (and science) of decision making that is making choices among desirable alternatives.
Early research, however, invested most of its energy in logical and mathematical models that studied decision making from the perspective of the theory of games and economics or that used statistical models to demonstrate decision biases. [33] Admittedly, this has been very important to evaluate and rank the alternatives of a decision, based on a set of goals and purposes, stakeholders and affected groups, many criteria, and their subcriteria. A lot of factors in decisions, however, may be intangible and have no measurements to serve as a guide to ordering alternatives and weighing the likely consequences of choosing each. [34] Numerical models, in many respects, were not enough to prioritize such factors so that they may be compared between themselves and with the other measurable factors. Not only does one need to create priorities for the alternative criterion variables, but also just for the criteria (and subcriteria) in terms of a higher goal or, in case they depend on the alternatives, then in terms of the alternatives themselves. That’s why, a more elaborate method was needed to translate qualitative phenomena into measurable numeric relations and, thus, to derive priority scales of absolute judgments that show, how much more, one element prevails over another with respect to the criterion or property to which they are compared. The analytic hierarchy process (AHP), which belongs to the Multi-criteria Decision-Making (MCDM) methods, turned out to be a widely plausible technique for that.

This study focuses on the introduction and application of the AHP method for the selection of decommissioning scenarios because:

1) a fundamental contention of this thesis is that decision making with regards to decommissioning scenarios is a joint function of two factors: (1) features of the activity, and (2) the decision maker’s knowledge and experience relevant to that activity;
2) through its comparisons and reliance on the judgments of specialists, AHP can derive relative priorities that measure intangibles in decision making at varying levels of hierarchy and complexity; and [34]
3) AHP has become increasingly international in scope in recent years and popular for research in many fields, including nuclear research and development and decommissioning projects. [35]

2.2.4 Methodology
Before anything, decision makers size up the situation, determine if a problem exists, and then, if so, think of how to decide and act upon it. The following steps present in brief outline how to break down the decision process and factors that affect it according to the AHP. [34]

1) Define the problem and the kind of knowledge sought.
2) Structure the decision hierarchy from the top with the goal of the decision then the objectives from a broad perspective, through the intermediate levels (criteria on which subsequent elements depend) to the lowest level (which is usually the set of alternatives).
3) Construct a set of comparison matrices. Each element in an upper level is used to compare the elements in the level immediately below it.
4) Use the priorities obtained from the comparisons to weigh the priorities in the level immediately below. Do this for every element. Then for each element in the level below, add its weighted values and obtain its overall (or global) priority. Continue this process of weighing and adding until the final priorities of the alternatives in the bottommost level are obtained.

Table (1) exhibits the scale the AHP uses to make comparisons. [34]

Table 1 The fundamental 1–9 scale of absolute numbers used to make comparisons with AHP.

<table>
<thead>
<tr>
<th>Intensity of importance (i over j)</th>
<th>Definition</th>
<th>Explanation</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Equal importance</td>
<td>Two activities contribute equally to the objective</td>
</tr>
<tr>
<td>2</td>
<td>Weak or slight importance</td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>Moderate importance</td>
<td>Experience and judgment slightly favor one activity over another</td>
</tr>
<tr>
<td>4</td>
<td>Moderate plus</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>Strong importance</td>
<td>Experience and judgment strongly favor one activity over another</td>
</tr>
<tr>
<td>6</td>
<td>Strong plus</td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>Very strong importance</td>
<td>An activity is favored very strongly over another; its dominance demonstrated in practice</td>
</tr>
<tr>
<td>8</td>
<td>Very, very strong importance</td>
<td></td>
</tr>
<tr>
<td>9</td>
<td>Extreme importance</td>
<td>The evidence favoring one activity over another is of the highest possible order of affirmation</td>
</tr>
<tr>
<td>Reciprocals of above</td>
<td>If activity i has one of the above numbers assigned to it when compared with activity j, then j has the reciprocal value when compared with i</td>
<td>A reasonable assumption</td>
</tr>
<tr>
<td>1.1-1.9</td>
<td>If the activities are very close</td>
<td>It may be difficult to assign the best value but when compared with other contrasting activities the size of the small numbers would not be too noticeable, yet they can still indicate the relative importance of the activities.</td>
</tr>
</tbody>
</table>

The AHP decomposes a complex MCDM problem into a system of hierarchies. The final step in the AHP deals with the structure of an $m \times n$ matrix where $m$ is the number of alternatives and $n$ is the number of criteria. The matrix is constructed by using the relative importance of the alternatives in terms of each criterion. The vector $a_{i1}, a_{i2}, a_{i3}, \ldots, a_{in}$ for each $i$ is the principal eigenvector of an $n \times n$ reciprocal matrix which is determined by comparisons of the impact of the $m$ alternatives on the $ith$ criterion.

The entry $a_{i1}$ in the $m \times n$ matrix represents the relative value of an alternative $A_i$ when it is considered in terms of a criterion $C_j$.

According to the AHP, the best alternative (in the maximization case) is indicated by the following relationship:
The importance of the AHP, its variants, and the use of comparisons in decision making is illustrated in the Results section.

3. General description of Oskarshamn Nuclear Power Plant

2.3 Introduction
In this chapter, Oskarshamn NPP is described along with its operational and physical characteristics, as well as components specific to the NPP’s third unit O3. This descriptive section will then be used to extract data necessary for dose rates modeling.

2.4 General description of the site
Oskarshamn NPP (shown in fig.8) is located directly on the peninsula of Simpevarp on the south-east coast of Sweden at the Baltic Proper (fig.9), 8 kilometers north-east of the village Figeholm with approximately 900 inhabitants and 20 kilometers north-east of the city Oskarshamn with approximately 26,300 inhabitants. [36] It has three boiling water reactors (BWRs), O1, O2, and O3 (shown in fig.10) of the same type ASEA-ATOM BWR, designed and built by Westinghouse Electric Sweden, but belonging to three different “reactor generations”. [36] The three reactors are owned and operated by the license holder OKG AB, and out of which only O3 is currently active and accounting for about 8.2% of the total electricity consumption of Sweden. [37] Oskarshamn NPP also has research laboratories and the Clab Interim Storage Facility for spent nuclear fuel, used by all NPPs in Sweden, owned and operated by the Swedish Nuclear Fuel and Waste Management Company AB, SKB and licensed by the Swedish Government. [36]

The supply of fresh water to the NPP is the Götemaren lake, located 8 kilometers north-west of Simpevarp. [38] The cooling water intake is a bottom intake through rock tunnels, 500 meters away from the shoreline, at a depth of 18 meters from the Baltic Sea, located on the southern part of the peninsula. [38] The cooling water discharges first into the 170,000 m² Hamnefjärden bay (fig.11). Because of the narrowness of Hamnefjärden, the flow rate of the discharged water is locally high (∼1 m/s) but decreases rapidly as the water returns to the Baltic Sea (after traveling about 800 meters). [38] Under full production operating conditions for O3, the temperature of the water at the outlet increases to 10 °C. [36]

The NPP’s buildings and their arrangement such that O1 and O2 are next to each other and O3 is situated separately, 500 meters northeast of O1 and O2, are as shown in table (2) and fig.12 and 13.
Figure 8 The three-unit Oskarshamn Nuclear Power Plant. (OKG, 2018)

Figure 9 The location of Oskarshamn NPP. (Lantmäteriet maps, 2018)
Figure 10 Oskarshamn nuclear power reactors O1, O2, and O3. (OKG, 2018)

Figure 11 The cooling water intake from the Baltic Sea and outlet into Hamnefjärden. (Lantmäteriet maps, 2018)
Table 2 Building designations for Oskarshamn NPP in relation to fig. 12.

<table>
<thead>
<tr>
<th>Building</th>
<th>O1</th>
<th>O2</th>
<th>O3</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor containment</td>
<td>RI</td>
<td>RI</td>
<td>A</td>
</tr>
<tr>
<td>Reactor building</td>
<td>R</td>
<td>R</td>
<td>B</td>
</tr>
<tr>
<td>Turbine and mid-section building</td>
<td>B</td>
<td>D</td>
<td>D</td>
</tr>
<tr>
<td>Auxiliary control building</td>
<td>E</td>
<td>H</td>
<td></td>
</tr>
<tr>
<td>Power and control building</td>
<td>E</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Auxiliary power building</td>
<td>N</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Office building and electric control building</td>
<td>M</td>
<td>E</td>
<td></td>
</tr>
<tr>
<td>Seawater cleaning building</td>
<td>R</td>
<td>F</td>
<td></td>
</tr>
<tr>
<td>New electric control building</td>
<td>T</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Yard</td>
<td>U</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Containment venting filter building</td>
<td>Y</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Active workshop</td>
<td>V</td>
<td>V</td>
<td>N</td>
</tr>
<tr>
<td>Waste treatment building</td>
<td></td>
<td>F</td>
<td></td>
</tr>
<tr>
<td>Cooling water pump building</td>
<td>J</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Diesel buildings</td>
<td>K</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Off-gas building</td>
<td>L</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Filtra building</td>
<td>M</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Entrance building</td>
<td>P</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Active culvert (underground)</td>
<td>Q</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Coolant intake building</td>
<td>R</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Service building</td>
<td>S</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Transformer building</td>
<td>T</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Gas storage</td>
<td>U</td>
<td></td>
<td></td>
</tr>
<tr>
<td>High voltage switchgear building</td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Condense cleanup system building</td>
<td>Z</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Figure 12 Building overview of Oskarshamn NPP. (OKG, 2018)
2.5 General description of how NPPs produce electricity

The schematic illustration of a nuclear power unit (Oskarshamn unit 3 in this case) and its electricity generation process appears in fig.14. It shows that the nuclear reactor itself is only one of many other components in such a power plant. The Rankine cycle is the process used by all large NPPs where the heat given off through nuclear fission is used to turn the water in the steam generators to high-pressure and high-temperature steam. [39] The latent energy of the steam is converted to mechanical energy by allowing the steam to expand against the blades of a turbine, and, hence, to turn the turbine shaft. [40] A large generator connected to the shaft continuously converts the mechanical turbine energy into electrical power that gets distributed through the electrical grid. [40] The re-condensation of the low-pressure steam leaving the turbine occurs in a steam condenser so that the resulting water is pumped back to the steam supply system. [40] This water-cooled condenser requires large quantities of ambient temperature cooling water which is usually obtained from artificial cooling ponds, cooling towers, rivers, or the ocean in this case. The nuclear steam supply system (NSSS) consists essentially of three major components: (a) a nuclear reactor supplying the fission heat energy, (b) several primary reactor coolant loops and pumps that circulate a coolant through the nuclear reactor to extract the fission heat energy, and (c) heat exchangers or steam generators that use the heated primary coolant to turn feedwater into steam generate the steam needed to drive the turbine-generator unit. [39]
2.6 The reactor core of a boiling water reactor

At the heart of the NSSS is the nuclear reactor. Far from being just a relatively simple “pile” of fuel and moderator, a modern nuclear reactor is an enormously complicated system designed to operate under the most severe conditions of temperature, pressure, and intense radiation. To introduce the general components of a typical nuclear reactor, below is a list of definitions related to the various components of the specific example of a BWR (fig.15).

**Fuel:** Any fissionable material which can be either fissile material such as $^{233}\text{U}$, $^{235}\text{U}$, $^{239}\text{Pu}$, or $^{241}\text{Pu}$ or fissionable material such as $^{232}\text{Th}$, $^{238}\text{U}$, or $^{240}\text{Pu}$. Most modern reactors use this fuel in a ceramic form, either as an oxide such as Uranium Dioxide ($\text{UO}_2$), a carbide such as Uranium Carbide ($\text{UC}$), or a nitride such as Uranium Mononitride ($\text{UN}$). The BWR uses enriched uranium with 3% uranium-235 as its fuel. This fuel is placed into the reactor in the form of uranium-oxide pellets in zirconium-alloy tubes. There may be as much as 140 tonnes of fuel in 75,000 fuel rods. Refueling a BWR involves removing the top of the reactor. The core itself is kept underwater, with the water shielding operators from radioactivity. Boron control rods enter the core from beneath the reactor.

**Fuel element:** The smallest sealed unit of fuel.

**Fuel assembly:** The smallest unit combining fuel elements into a bundle. In a BWR, fuel assemblies have a duct wall to prevent vapor radial drifting. Fuel is usually loaded into a reactor core or replaced one fuel assembly at a time.

**Moderator:** Material of low mass number which is inserted into the reactor to slow down or moderate neutrons via scattering collisions. Typical moderators include light water, heavy water, graphite, and beryllium. The BWR uses ordinary water (light water) as both its coolant and its moderator.
**Coolant:** A fluid that circulates through the reactor removing fission heat. The coolant can be either liquid (water or sodium) or gaseous (helium or carbon dioxide).

**Coolant channel:** One of the channels through which the coolant flows in the fuel lattice.

**Structure:** The geometry and integrity of the reactor core are maintained by sutural elements such as support plates, spacer grids, or metallic tubes used to clad the fuel in some reactor designs.

**Control elements:** Absorbing material inserted into the reactor to control core multiplication. Although most commonly regarded as movable rods of the absorber, control elements may also consist of fixed absorbers or absorbing materials dissolved in the coolant. Common absorbing materials include boron, cadmium, gadolinium, and hafnium.

**Reactor core:** The total array of fuel, moderator, and control elements. (fig. 15 and 16)

**Reflector:** Material that is characterized by a low absorption cross-section used to surround the core to reflect or scatter leaking neutrons back into the core.

**Shielding:** The reactor is an intense source of radiation. Not only must operating personnel and the public be shielded from this radiation, but reactor components must as well be protected. Therefore, absorbing material is introduced to attenuate both neutron and gamma radiation. Thermal shielding is used to attenuate the emergent core radiation to levels that do not result in significant heat generation and so, damage in the reactor’s components. Biological shielding reduces the radiation still further to acceptable levels for operating personnel.

**Support structure:** The support plates that serve to maintain the core geometry.

**Reactor pressure vessel:** The high-pressure containment for the reactor and associated primary coolant system.

---

*Figure 15 The reactor core 03. (OKG, 2018)*
O3 is the NPP’s reactor that is still in operation nowadays, one of the world’s largest BWRs with a gross electrical output of 1,450 MW, corresponding to 131.8% power (compared to design net capacity of 1100 MW ≡ 100%), and a 'resistor-capacitor' (RC) flow "window" ranging between 14,350 kg/s (90.8% flow) and 15,800 kg/s (100% flow) at 131.8% power. The reactor core, rising to 4 meters in height, is formed by a cylindrical arrangement of 700 fuel assemblies, out of which 175 ≡ 25% are changed during the annual outage time. The reactor is controlled by 169 fine-motion control rods adjusted by motors and the reactor main circulation is maintained by 8 speed-controlled reactor internal pumps (RIPs). The reactor core is surrounded by the core shroud, a thick cylindrical stainless-steel assembly mounted by the core grid. The reactor containment, the last barrier for radioactive releases to the environment, is a multi-compartment, multi-floor, and cylindrical shaped building with a dome at the top. The O3 containment has a design pressure of 600 kPa and is blown out through pressurized nitrogen (Ne-inlet connection). Filtered venting of the containment, necessary to prevent the occurrence of core meltdown accidents, is achieved by the Multi Venturi Scrubber System (MVSS), directly connected to the upper part of the containment. The turbine building is the largest of buildings, holding a double axial high-pressure turbine and three double axial low-pressure turbines. The hydrogen-cooled rotor and the water-cooled stator are coupled to the same shaft with the generator which in turn is connected to the national 400-kV electrical grid that provides power to the NPP during a shutdown, along with the regional 130-kV grid. The safety classification of the electrical systems divides them into four physically and functionally distinct subsystems (A, B, C, and D), each with its own emergency diesel generator.

2.8 Main data
Table (3) presents the main technical data for O3, and table (9) presents the radioactivity data for the reactor pressure vessel (RPV) and the internal parts of O3 assuming a 60-year reactor lifetime.
### Table 3 Main technical data for O3. (SKB, 2013)

<table>
<thead>
<tr>
<th></th>
<th>Unit</th>
<th>O3</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor</td>
<td></td>
<td>ASEA-ATOM</td>
</tr>
<tr>
<td>Turbine</td>
<td></td>
<td>Alstom</td>
</tr>
<tr>
<td>Construction start</td>
<td></td>
<td>May 1980</td>
</tr>
<tr>
<td>Commission</td>
<td></td>
<td>March 1985</td>
</tr>
<tr>
<td>Decommissioning start</td>
<td></td>
<td>2035 or 2045</td>
</tr>
<tr>
<td>Reactor building</td>
<td>m³</td>
<td>148,000</td>
</tr>
<tr>
<td>Turbine building</td>
<td>m³</td>
<td>275,000</td>
</tr>
<tr>
<td>Total construction volume</td>
<td>m³</td>
<td>840,000</td>
</tr>
<tr>
<td>Rockblasting</td>
<td>m³</td>
<td>745,000</td>
</tr>
<tr>
<td>Formwork</td>
<td>m³</td>
<td>512,000</td>
</tr>
<tr>
<td>Concrete</td>
<td>m³</td>
<td>126,000</td>
</tr>
<tr>
<td>Reinforcement</td>
<td>tonnes</td>
<td>17,000</td>
</tr>
<tr>
<td>The total height of the reactor building</td>
<td>m</td>
<td>64</td>
</tr>
<tr>
<td>Height above ground level</td>
<td>m</td>
<td>57</td>
</tr>
<tr>
<td>Stack height</td>
<td>m</td>
<td>103</td>
</tr>
<tr>
<td>Thermal reactor power</td>
<td>MW</td>
<td>3,900</td>
</tr>
<tr>
<td>Reactor operating pressure</td>
<td>MPa</td>
<td>7</td>
</tr>
<tr>
<td>Reactor steam temperature</td>
<td>°C</td>
<td>286</td>
</tr>
<tr>
<td>Steam flow</td>
<td>Kg/s</td>
<td>2.115</td>
</tr>
<tr>
<td>Inner height</td>
<td>m</td>
<td>21.1</td>
</tr>
<tr>
<td>Outer height</td>
<td>m</td>
<td>21.4</td>
</tr>
<tr>
<td>Inner diameter</td>
<td>m</td>
<td>6.43</td>
</tr>
<tr>
<td>Outer diameter</td>
<td>m</td>
<td>6.75</td>
</tr>
<tr>
<td>Wall thickness</td>
<td>mm</td>
<td>156</td>
</tr>
<tr>
<td>Weight with head</td>
<td>tonne</td>
<td>760</td>
</tr>
<tr>
<td>Absorber material</td>
<td></td>
<td>B$_4$C</td>
</tr>
<tr>
<td>Number of control rods (cruciform)</td>
<td>No. of units</td>
<td>169</td>
</tr>
<tr>
<td>Electrohydraulic drive mechanism</td>
<td></td>
<td>169</td>
</tr>
<tr>
<td>Number</td>
<td>No. of units</td>
<td>8</td>
</tr>
<tr>
<td>Maximum flowrate per pump</td>
<td>M³/s</td>
<td>1.860</td>
</tr>
<tr>
<td>Pressure</td>
<td>MPa</td>
<td>0.39</td>
</tr>
<tr>
<td>Rated power</td>
<td>MWe</td>
<td>1,465</td>
</tr>
<tr>
<td>Fuel</td>
<td></td>
<td>SVEA-96</td>
</tr>
<tr>
<td>Number of fuel assemblies</td>
<td>No. of units</td>
<td>700</td>
</tr>
<tr>
<td>Number of fuel rods per assembly</td>
<td>No. of units</td>
<td>96/100</td>
</tr>
<tr>
<td>Cladding material</td>
<td></td>
<td>Zr-2</td>
</tr>
</tbody>
</table>

### Table 4 Neutron-induced activity in the Reactor Pressure Vessel and in the internal parts of O3 (one-year decay time). (Jonasson, 2012)

<table>
<thead>
<tr>
<th>Part</th>
<th>Reactor Containment Contaminated Concrete</th>
<th>Reactor Containment Reinforcement</th>
<th>Reactor Pressure Vessel (RPV)</th>
<th>RPV Insulation</th>
<th>Core Frame</th>
<th>SRM/WRNM detectors</th>
<th>TIP-detector</th>
</tr>
</thead>
<tbody>
<tr>
<td>Active weight (tonne)</td>
<td>600.0</td>
<td>56.2</td>
<td>760</td>
<td>6.0</td>
<td>82.0</td>
<td>2.4</td>
<td>0.0</td>
</tr>
<tr>
<td>Nuclide</td>
<td>Bq</td>
<td>Bq</td>
<td>Bq</td>
<td>Bq</td>
<td>Bq</td>
<td>Bq</td>
<td>Bq</td>
</tr>
<tr>
<td>H-3</td>
<td>6.9e+11</td>
<td>5.4e+11</td>
<td>1.0e+03</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>C-14</td>
<td>2.1e+08</td>
<td>2.7e+07</td>
<td>4.6e+08</td>
<td>2.2e+07</td>
<td>9.9e+12</td>
<td>1.2e+11</td>
<td>1.7e+05</td>
</tr>
<tr>
<td>Co-60</td>
<td>2.3e+10</td>
<td>3.6e+10</td>
<td>4.1e+11</td>
<td>8.1e+10</td>
<td>4.8e+15</td>
<td>1.1e+15</td>
<td>1.7e+09</td>
</tr>
<tr>
<td>Fe-55</td>
<td>7.8e+10</td>
<td>5.9e+11</td>
<td>5.2e+12</td>
<td>8.3e+10</td>
<td>3.7e+16</td>
<td>4.0e+15</td>
<td>2.9e+10</td>
</tr>
<tr>
<td>Ni-63</td>
<td>1.7e+08</td>
<td>5.2e+09</td>
<td>2.2e+11</td>
<td>1.7e+10</td>
<td>8.1e+15</td>
<td>1.1e+14</td>
<td>5.6e+07</td>
</tr>
<tr>
<td>Sr-90</td>
<td>2.8e+06</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1.0e+10</td>
<td>1.9e+05</td>
</tr>
</tbody>
</table>
4. The four potential post-decommissioning scenarios

This section is a site-specific application of a probabilistic dose analysis, characterized by radionuclides of interest, probabilistic parameters, and most likely release scenarios for O3.

When considering decommissioning requirements, it is desirable to achieve the immediate dismantling of the nuclear power plant and the subsequent release of the site from some or all of the regulatory controls, while ensuring public and worker safety and protecting the environment from the harmful effects of ionizing radiation. [44] A strong case can be made that hazards associated with nuclear power, which could be justified while the plant is supplying power for as long as required, are hard to justify when it stops providing any substantial gainful activity at the site. This makes it clear why, and as previously was mentioned, the ultimate objective of decommissioning is not only the release but the release for unrestricted or specific profitable uses of the site. Demonstrating (and eventually receiving) the benefits a decommissioned site will produce strengthens the expenditure on decommissioning and the disposal of radioactive waste to be supported and possibly reduced and shows how and where both financial and non-financial payback can occur and, at best, meet public expectations. That’s why the definition of decommissioning was further developed by the IAEA [44] to become as follows: “Decommissioning is the administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility and to restore the site to new uses.” All this meant one thing: a circular scheme of the nuclear lifecycle as depicted in fig.17 (as opposed to the one in fig.2).

But is post-decommissioning redevelopment/reuse always the best decision? And how do we know? Radiological criteria for restricted or unrestricted release of sites/facilities have an important impact on redevelopment and reuse options. [45] In this connection, a detailed radiation survey is performed, and the total amount of activity on the site is calculated,
since the beginning of the planning stage of decommissioning, where normally redevelopment and reuse opportunities should be considered. [46] These calculations are beyond the scope of this study, and, thus, sufficient information on the neutron-induced activity\textsuperscript{12} and surface contamination levels in the reactor containment concrete of O3, which have arisen during operation and are projected to remain at shutdown, is obtained from the available operating history. [43] Highlighting the uncertainty in the data is essential to address its impact on the results.

Knowledge of the radionuclide inventory is important for predicting the rates of radioactive decay following the shutdown and, thus, predicting the redevelopment and reuse potential of the facility and/or site at each forecast time step. Following on from these periods is an end-point phase where hazards and risk consequences are reduced to permissible levels for release. [47] In Sweden, the NPP is released according to the Swedish Radiation Safety Authority (SSM) requirements for the protection of human health and the environment in connection with discharges of radioactive substances and will be subsequently exempted from the requirements of the Nuclear Activities Act. [48] In fact, SKB assumes in its decommissioning studies that, after their release, the NPP’s area and buildings will be used in some way for new industrial activities. Despite adopting the system of general clearance in Sweden, this study uses the alternative approach of multi-pathway exposure source-to-dose modeling to derive site-specific release criteria. General clearance requires the regulatory body to develop generic release criteria (as recommended by international organizations) for the operator to use; however, the operator can opt to derive site-specific release criteria that the regulatory body should approve. The first approach binds the operator to demonstrate compliance with the generic release criteria, but which might be too conservative due to the general assumptions made for performing the dose assessment. This could result in higher costs than necessary for more extensive cleanups. [22] While the site-specific approach requires extensive knowledge of the situation and extra work from both the operator and the regulator, it is likely to result in a less stringent set of release criteria and a faster and more cost-effective cleanup. [22] Another reason why it is best to use site-specific DCGLs is that, normally, facilities and sites are assessed on a case-by-case basis when it comes to potential uses of the site. RESRAD-BUILD is used to calculate release levels for some radionuclides which are typical of those found in BWR plants and are deemed to be relevant for the release measurements according to this study: H-3, C-14, Co-60, Fe-55, Ni-63, Sr-90, Cs-134, and Cs-137.

This section is made with boundary conditions as follows, see table (5).

\textit{Table 5 Decommissioning plan boundary conditions for Oskarshamn reactor O3.}

<table>
<thead>
<tr>
<th>Boundary conditions</th>
<th>Oskarshamn unit 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>Decommissioning strategy</td>
<td>Direct dismantling is the chosen option.</td>
</tr>
<tr>
<td></td>
<td>No period for reestablishment is needed.</td>
</tr>
</tbody>
</table>

\textsuperscript{12} Note that, in this study, the long-lived radionuclides include fission and activation products and are assumed to be found, in considerable concentrations, solely in the reactor internals close to the core.
Decommissioning schedule

- The starting point is one year before shutdown (2044).
- The total defueling time is 2 to 3 years.
- The shutdown operation and preparations for dismantling take 2 years.
- The dismantling period is 5 to 6 years and includes free release measurements.
- The start-to-finish demolition period (if applicable) for all the demolition activities is projected to be about 10 months.

Radiological criteria

- Site characterization is limited to a radiation survey of the reactor’s major component, the reactor pressure vessel, and its internal parts.

- The radiation dose shall not exceed 0.1 mSv/year to an individual in the critical group.

SSMFS 1996:2 The Swedish Radiation Safety Authority’s Regulations on Clearance of Goods and Oil from Nuclear Facilities.
- The limit for free release is 500 Bq/kg.

Contaminated soil

- Contaminated soil has not been identified at O3.

Hazardous waste

- No hazardous waste is evaluated in the study.
- No asbestos is assumed to be present in O3.

Within strictly defined limits, site-specific parameter values are supplied under the assumptions that (1) the reactor building is decommissioned in 10 years and, if applicable, demolished within a year and that (2) the whole area is contaminated at the same level with no dilution by other non-radioactive materials. The main input parameters for executing the RESRAD computer code, and corresponding to each hypothetical future use scenario for the reactor building, are found in the table (11).

Table 11 Input parameters for a one-room model in RESRAD-BUILD computer code.

<table>
<thead>
<tr>
<th>Time parameters</th>
<th>Release</th>
<th>Dose conversion factors</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exposure duration (d)</td>
<td>101</td>
<td>ICRP-60</td>
</tr>
</tbody>
</table>

13 For the sake of simplicity, it may be appropriate to assume numerical equivalence between levels expressed in Bq/Kg and Bq/m².

14 Shearing demolition because it is less labor intensive and takes less time than non-shearing demolition scenario based on use of a wire saw.
### Indoor fraction (50th percentile range)

<table>
<thead>
<tr>
<th></th>
<th>0.33</th>
<th>0.194 – 0.365</th>
<th>0.194 – 0.365</th>
<th>0.625 – 0.729</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of times for calculation (unitless)</td>
<td>3</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>User-specified times for dose calculations (yr)</td>
<td>$t_0 = 0$; $t = 0.28$; $t = 0.55$</td>
<td>$t_0 = 0$; $t = 1$; $t = 10$</td>
<td>$t_0 = 0$; $t = 10$; $t = 1000$</td>
<td></td>
</tr>
<tr>
<td>Maximum time integration points for Dose/Risk (unitless)</td>
<td>1 and 2</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

### Building Parameters

#### Loguniform distribution

<table>
<thead>
<tr>
<th>Deposition velocity (m/s)</th>
<th>Min</th>
<th>Max</th>
<th>Min</th>
<th>Max</th>
<th>Min</th>
<th>Max</th>
<th>Min</th>
<th>Max</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>2.7e-6</td>
<td>2.7e-3</td>
<td>2.7e-6</td>
<td>2.7e-3</td>
<td>2.7e-6</td>
<td>2.7e-3</td>
<td>2.7e-6</td>
<td>2.7e-3</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Resuspension rate (s⁻¹)</th>
<th>Min</th>
<th>Max</th>
<th>Min</th>
<th>Max</th>
<th>Min</th>
<th>Max</th>
<th>Min</th>
<th>Max</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>5.1e-10</td>
<td>5.1e-7</td>
<td>1.4e-6</td>
<td>2.0e-6</td>
<td>1.9e-9</td>
<td>1.9e-6</td>
<td>7.7e-7</td>
<td>1.1e-6</td>
</tr>
</tbody>
</table>

| Building air exchange rate (h⁻¹) | 6 (49) | 4 (49) (area w/ a risk of explosive atmosphere) | 7 (49) | 0.27 (50) (energy-efficient houses) |

### Receptor parameters

#### Number of receptors (unitless)

|                | 3    |               |               |               |

#### Time fraction (unitless)

|                | 1³      |               |               |               |

#### Inhalation rate (m³/d)

<table>
<thead>
<tr>
<th>Short-term heavy activities</th>
<th>Long-term moderate to heavy activities</th>
<th>Long-term light activities</th>
<th>Typical indoor activity levels¹⁶</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>25.6 (51)</td>
<td>33.6 (52)</td>
<td>23 (53)</td>
<td>15.2 (51)</td>
</tr>
</tbody>
</table>

#### Indirect ingestion rate¹⁷ (m²/h)

<table>
<thead>
<tr>
<th></th>
<th>High ingestion rate estimate: 7.5e-3 (52)</th>
<th>Low ingestion rate estimate: 7.5e-5 (52)</th>
</tr>
</thead>
</table>

### Source parameters

#### Number of sources (unitless)

|                | 1    |               |               |               |

#### Source type (unitless)

|                | Area source |               |               |               |

#### Source area (m²)

|               | The RPV area: 525.37 |               |               |               |

#### Source location (m)

|                     | x: 1.8, y: 3.4, z: 10.7 |               |               |               |

#### Air release fraction (unitless)

<table>
<thead>
<tr>
<th></th>
<th>Minimum</th>
<th>Maximum</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>1e-6</td>
<td>1e-6</td>
<td>1</td>
<td>1</td>
<td></td>
</tr>
</tbody>
</table>

#### Removable fraction (unitless)

|                | 1       | 0.1     | 0.1           | 0.01           |

#### Radionuclide concentration/activity in the reactor containment concrete (Bq) obtained from ALARIA Engineering (Jonasson, 2012)

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>6.9e+11</th>
<th>2.1e+08</th>
<th>2.3e+10</th>
<th>7.8e+10</th>
</tr>
</thead>
<tbody>
<tr>
<td>H-3</td>
<td>6.9e+11</td>
<td>2.1e+08</td>
<td>2.3e+10</td>
<td>7.8e+10</td>
</tr>
<tr>
<td>C-14</td>
<td>2.1e+08</td>
<td>2.3e+10</td>
<td>7.8e+10</td>
<td>1.7e+08</td>
</tr>
<tr>
<td>Co-60</td>
<td>2.3e+10</td>
<td>1.7e+08</td>
<td>2.8e+06</td>
<td>2.2e+09</td>
</tr>
<tr>
<td>Fe-55</td>
<td>7.8e+10</td>
<td>2.8e+06</td>
<td>2.2e+09</td>
<td>3.0e+06</td>
</tr>
<tr>
<td>Ni-63</td>
<td>1.7e+08</td>
<td>2.8e+06</td>
<td>2.2e+09</td>
<td>3.0e+06</td>
</tr>
<tr>
<td>Sr-90</td>
<td>2.8e+06</td>
<td>2.2e+09</td>
<td>3.0e+06</td>
<td></td>
</tr>
<tr>
<td>Cs-134</td>
<td>2.2e+09</td>
<td>3.0e+06</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cs-137</td>
<td>3.0e+06</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

---

¹⁵ Because the receptor is assumed to be always inside the same room while inside the building.


¹⁷ The receptor does not need to be in the same room as the source to be exposed by this pathway.
<table>
<thead>
<tr>
<th>Shielding parameters</th>
<th>Shielding material</th>
<th>The RPV wall</th>
<th>A new wall</th>
<th>No shielding</th>
</tr>
</thead>
<tbody>
<tr>
<td>Shielding thickness (cm)</td>
<td>Heavyweight concrete</td>
<td>15.6</td>
<td>30 (average concrete thickness)</td>
<td>0 (the minimum and most probable value in the triangular distribution assumed in RESRAD)</td>
</tr>
<tr>
<td>Shielding density (g/cm$^3$)</td>
<td></td>
<td>3.51$^{18}$ (55)</td>
<td>2.3 (56)</td>
<td>0.001225$^{19}$</td>
</tr>
</tbody>
</table>

The EPA’s Exposure Factors Handbook [51] provides a comprehensive summary of available data on various factors used in exposure assessments, including human activity patterns, time spent at work, and time spent indoors at a residence, and this is where from indoor fraction values came in.

As with the starting of dose calculations, up to nine user-specified times may be entered and can be integer years or fractions of years. The code automatically calculates the dose at time zero and the doses at each assessment time over the specified exposure duration. 1- and 10-year dose evaluations for the 365-day exposure duration produce doses incurred during the zeroth, first, and tenth year. The 10-month assessment time for demolition may be divided into three equal intervals of 101 days each. The exposure duration is then entered as “101 days” and the number of evaluations as “3 times”. Doses are calculated at $t = 0$, $t = 0.28$, and $t = 0.55$ years over the first, second, and third periods respectively. The total dose to each receptor is the sum of the individual doses calculated at each evaluation time, which, in their turn, are obtained by numerically integrating the dose rate over the exposure duration. The dosimetry used is based on ICRP 60.

Integration starts with points 1 and 2. The results are compared with one another to check whether they agree sufficiently, i.e. within 1%. If they do, no further integration is needed, but if not, results can be improved upon with one additional integration point, and the results are then compared with those obtained using two integration points. If the difference between them is less than 1%, the results obtained using three integration points are adopted. However, a lack of a 1% agreement would require further integration. The number of integration points is then increased from 3, $2n + 1$ sequence wise, to a maximum of 257. Integrating and assessing agreement continue until either the user-specified maximum integration point is reached or a better than 1% convergence is achieved. The last integration results are the ones finally adopted.

---

$^{18}$ This value also falls within the range of densities suggested by Merritt et al. (56) for heavyweight concrete.

$^{19}$ According to assumptions made by the International Standard Atmosphere (ISA) model.
Loguniform distributions were examined from several sources to represent the indoor resuspension rates for the demolition setting [57], nuclear occupational setting [58], non-nuclear occupational setting [57], and residential setting [59].

For DCGLs derivation, the peak mean dose at each evaluation time is obtained, and the highest value among them is selected and compared with the dose limit i.e. 0.1 mSv/yr. The peak dose from the demolition worker is taken at the end of the demolition period i.e. t = 0.83 years. Assuming a 10-year decommissioning period, the peak dose from the industrial worker is taken at t = 10 years after completion of decommissioning. The selected timeframe is also based on EPA’s recommendation that hypothetical future scenarios for nuclear sites should address potential site uses within the next 10 years, and not necessarily beyond that. (60) Noting that, for the residential scenario solely, the radiological risk associated with long-term exposure will be quantified up to 1000 years. The dose to the domestic resident will be taken in the 10th year to be compared with that of the industrial worker, but the peak dose is captured at t = 1000 years. The highest dose among the selected ones from each period is taken to calculate DCGLs which are derived as follows:

\[ DCGL_i = \frac{\text{Dose limit}_i}{\text{DSR}_i} \]  \( (35) \)

Where \( DCGL_i \) = derived concentration guideline level of radionuclide \( i \) (Bq/m\(^2\))

\( \text{Dose limit}_i \) = allowable dose of radionuclide \( i \) (mSv/yr)

\( \text{DSR}_i \) = dose to source ratio (mSv/yr per Bq/m\(^2\))

The gross DCGL for surfaces is calculated using the following equation:

\[ DCGL_{gross} = \frac{1}{\left( \frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \cdots + \frac{f_n}{DCGL_n} \right)} \]  \( (36) \)

Where \( f_i \) is the fraction of activity contributed by each radionuclide.

5. Results

5.1 Radiological modeling results

Dose trends for each of the 8 radionuclides considered in this study, represented by the 4 different dose assessment scenarios are shown in the graphs of fig.18-40. These are collective doses i.e. the sum of the individual doses received in a given period by population from exposure to a specified source of radiation. The overall results show that Co-60 determines the dose rate on all activated BWR materials, with a smaller contribution from Cs-134, and that the other radionuclides have considerably less restrictive clearance levels.
Figure 18 Collective dose from exposure to Cs-137 (mSv/yr) in the demolition and industrial scenarios.

Figure 19 Collective dose from exposure to Cs-137 (mSv/yr) in the residential scenario.

Figure 20 Collective dose from exposure to Cs-134 (mSv/yr) in the demolition, nuclear, and non-nuclear scenarios.
Figure 21 Collective dose from exposure to Cs-134 (mSv/yr) in the residential scenario.

Figure 22 Collective dose from exposure to Sr-90 (mSv/yr) in the demolition and nuclear scenarios.

Figure 23 Collective dose from exposure to Sr-90 (mSv/yr) in the non-nuclear scenario.
Figure 24 Collective dose from exposure to Sr-90 (mSv/yr) in the residential scenario.

Figure 25 Collective dose from exposure to Ni-63 (mSv/yr) in the demolition and nuclear scenarios.

Figure 26 Collective dose from exposure to Ni-63 (mSv/yr) in the non-nuclear scenario.
Figure 27 Collective dose from exposure to Ni-63 (mSv/yr) in the residential scenario.

Figure 28 Collective dose from exposure to Co-60 (mSv/yr) in the demolition, nuclear, and non-nuclear scenarios.

Figure 29 Collective dose from exposure to Co-60 (mSv/yr) in the residential scenario.
Collective dose from exposure to Fe-55 (mSv/yr) in the demolition and nuclear scenarios.

Collective dose from exposure to Fe-55 (mSv/yr) in the non-nuclear scenario.

Collective dose from exposure to Fe-55 (mSv/yr) in the residential scenario.
Figure 33 Collective dose from exposure to C-14 (mSv/yr) in the demolition and nuclear scenarios.

Figure 34 Collective dose from exposure to C-14 (mSv/yr) in the non-nuclear scenario.

Figure 35 Collective dose from exposure to C-14 (mSv/yr) in the residential scenario.
Figure 36: Collective dose from exposure to H-3 (mSv/yr) in the demolition and nuclear scenarios.

Figure 37: Collective dose from exposure to H-3 (mSv/yr) in the non-nuclear scenario.

Figure 38: Collective dose from exposure to H-3 (mSv/yr) in the residential scenario.
Figure 39 Total exposure (mSv/yr) in the demolition, nuclear, and non-nuclear scenarios.

Figure 40 Total exposure (mSv/yr) in the residential scenario.

The lowest levels for all radionuclides were found in the short-term demolition scenario followed by values from the long-term non-nuclear scenario, and then nuclear scenario, with no major differences between these two for Cs-137, Cs-134, and Co-60. While the highest doses by nuclides, as expected, were observed in the long-term residential scenario. The difference between residential levels and levels from the other scenarios principally follows that in parameters, specifically in the indoor fraction, the building air exchange rate, the inhalation rate, and the shielding material.

At the beginning of the exposure period, the initial dose was the highest for all scenarios because the studied radionuclides contribute most of their dose via the direct exposure pathway, the dominant exposure pathway at that time. The contaminants gradually start spreading far from their point of release, and the principal exposure pathway changes from direct to inhalation, which best explains the slow decrease in dose during the first 10 months. By the end of the first year, the collective dose from the demolition scenario...
became zero because demolition will have finished, and workers will have left the site. The same happened to the individual doses of Ni-63, Fe-55, and H-3.

The result for Ni-63 in all four scenarios concurs well with the fact that nickel isotopes can only be found in small amounts in the air following the decommissioning of a nuclear power plant. [61]

The dose values from Fe-55 in both the demolition and nuclear scenarios were about 4 to 5 orders of magnitude lower than those in the non-nuclear one, and 5 to 6 orders of magnitude lower than those in the residential one. Although few studies have been put forward about the behavior of iron-55 in reactor decommissioning situations, the radionuclide's activity levels were shown to be small as a result of low particle settling rates in comparison with its physical half-life that is 2.7 years. [62] Its contribution to the dose is therefore comparatively low, which supports the result for Fe-55.

Nevertheless, in the earliest stage post-shutdown, non-gamma activity is dominated by Ni-63 in neutron-irradiated stainless steels and by Fe-55 in carbon steels. [62]

Also, the result for H-3 in the demolition and nuclear scenarios is in complete agreement with IAEA’s findings [63] that state that the contribution from tritium to the total dose is known to be very small and implies that any risk from tritium alone would be negligible.

In the early months of the assessment period, the dose for C-14 remained almost constant, then disappeared at the beginning of year one in all scenarios except in the residential one, wherein 1000 years later, the dose was 1.60E-02 mSv with a slight change from the baseline 5.03E-02 mSv at time zero. Although the Nuclear Waste Technical Review Board (NWTRB) [64] affirms that annual doses predicted from C-14 releases would be a “very small fraction” of the total dose being received by workers during actual decommissioning operations, C-14 remains of particular importance because of its long half-life and, hence, contribution to long-term individual and public radiation doses.

As a further explanation for these four results, low-energy beta-emitting radionuclides generally result in low doses. Although its contribution to the radiation dose is usually considered to be unimportant compared to that of other pathways, percutaneous absorption is especially important for beta emitters (H-3 and C-14 of special concern). But because the skin dose is not accounted for by RESRAD, these radionuclides exhibited lower doses than usual (fig.33-38).

However, this result shouldn’t be viewed as a constraint for two reasons:

1) there is always an expected relationship between weak beta-emitters and other radionuclides at a given time in their production cycle. For example, Ni-63 and Fe-55 can often be correlated to Co-60, and Sr-90 to Cs-134, both of which are major contributors to the dose and whose concentrations can be used as an alternative to estimate those of the weaker radionuclides (when modeled ones are far too low) especially in the occupational scenarios; [63] and
2) it was still possible to properly derive their DCGLs from other scenarios as demonstrated later, see table (6).

It is clear by now that the radiological properties of Co-60 are so dominant that its decay governs the reduction in gamma dose rates over the entire assessment period. This is in perfect accordance with the evidence showing that the activated corrosion products, especially Co-60, dominate the radiation levels around most of the reactor process systems after shutdown, [42] with values starting at 2.24E+03, 8.95E+03, 6.66E+03, and 2.28E+05 in the first, second, third and fourth scenarios respectively, and ending at 2.16E+03 and 1.58E+03 in scenarios 2.1 and 2.2 by the end of the first assessment period, and at 5.17E-01 by the end of the 1000-year assessment period for the third scenario.

For the occupational exposure, the only radionuclide of importance in the carbon and reinforcement steels of the RPV wall, apart from Co-60, is Cs-134. [65] This supports the relatively high levels of Cs-134 in all scenarios, with values starting at 7.20E+01, 2.88E+02, 2.13E+02, and 1.38E+04 in the first, second, third, and fourth scenarios respectively, and ending at 8.95E+00 and 6.44E+00 in scenarios 2.1 and 2.2 by the end of the first assessment period, and at 3.83E-08 by the end of the 1000-year assessment period for the third scenario.

The assumed presence of cesium in the containment can be attributed to a process called revaporization. When decay heating of deposits in the reactor coolant system produces temperatures high enough, revaporization happens and causes the prolonged, low-intensity release of cesium, being a volatile radionuclide, to the containment. [66] In their recent study, Tuca et al. [67] explain that the values for Cs-134 are usually derived from those for Cs-137 because the latter would dominate the mix. But as against this assessment, the result for Cs-137 here shows that it decayed to levels much lower than those of Cs-134. Apart from this disagreement, the result is aligned with previous results reported in the literature and that show that the activity of Cs-137, being a long-lived radionuclide, decreases slowly with time. [66] The relatively low levels in the demolition and both industrial scenarios are also supported by previous evidence that (1) only small quantities of Cs-137 can be found in the air due to nuclear activities, [68] (2) cesium compounds can travel long distances in the air, [68] and (3) Cs-137 is rapidly and strongly adsorbed by fine soil particles, something RESRAD-ONSITE not -BUILD would calculate. [69]

Finally, although the levels of airborne Sr-90 were found not to be “extremely small” as indicated by NRC, [70] radiation doses from it were indeed “a tiny fraction of less than 0.01 mSv” by the end of all three assessment periods.

A closer look at the main assessment period of 10 years, common to the rest of the scenarios, shows that the dose from the industrial ones is lower than that from the residential one of the same times.

The average annual dose rate in the non-nuclear scenario was, in some instances, higher than in the nuclear one because some non-nuclear technologies and industrial activities enhance the environmental levels of natural radionuclides such as fossil fuel combustion.
and production or use of phosphate-rock products. Another factor is the adequate precautionary measures workers take and the protective clothing and equipment wear on nuclear sites.

The results show the dose trends over the years for each radionuclide from the four scenarios. But because the endpoint for decontamination should be clearly defined in terms of bulk activity (Bq/g) or surface activity (Bq/m²), these doses were converted to surface-activity concentration using equations (35) and (36). By this means, it was possible to obtain nuclide-specific values of clearance levels and total clearance levels for each scenario.

Table (12) shows the dose for each period of all four scenarios, the maximum dose obtained, and the calculated DCGLs for the 8 radionuclides.

Table 12 The site-specific DCGLs for the eight radionuclides considered in the study.

<table>
<thead>
<tr>
<th>Radionuclides</th>
<th>Clearance</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Complete demolition (short-term)</td>
</tr>
<tr>
<td></td>
<td>Reuse of building (long-term)</td>
</tr>
<tr>
<td></td>
<td>Unrestricted use</td>
</tr>
<tr>
<td></td>
<td>Restricted use</td>
</tr>
<tr>
<td></td>
<td>Unrestricted use</td>
</tr>
<tr>
<td>Scenario 1</td>
<td>Scenario 2.1</td>
</tr>
<tr>
<td>Dose of</td>
<td>Dose of nuclear industrial worker</td>
</tr>
<tr>
<td>demolition</td>
<td>Dose of non-nuclear industrial worker</td>
</tr>
<tr>
<td>worker</td>
<td>Dose of residential farmer</td>
</tr>
<tr>
<td>(mSv/yr)/</td>
<td>(mSv/yr)/(Bq/m²)</td>
</tr>
<tr>
<td>(Bq/m²)</td>
<td>(mSv/yr)/(Bq/m²)</td>
</tr>
<tr>
<td>Max</td>
<td>DCGL</td>
</tr>
<tr>
<td>Assessment</td>
<td>at 0.83</td>
</tr>
<tr>
<td>period</td>
<td>at 10</td>
</tr>
<tr>
<td>Nuclide</td>
<td>at 10</td>
</tr>
<tr>
<td></td>
<td>at 100</td>
</tr>
<tr>
<td>Cs-137</td>
<td>5.01e-03</td>
</tr>
<tr>
<td>Cs-134</td>
<td>9.22e+00</td>
</tr>
<tr>
<td>Sr-90</td>
<td>8.58e-06</td>
</tr>
<tr>
<td>Ni-63</td>
<td>1.67e-07</td>
</tr>
<tr>
<td>Co-60</td>
<td>3.40e+02</td>
</tr>
<tr>
<td>Fe-55</td>
<td>1.23e-05</td>
</tr>
<tr>
<td>C-14</td>
<td>7.78e-07</td>
</tr>
<tr>
<td>H-3</td>
<td>1.93e-05</td>
</tr>
</tbody>
</table>

The results are compared with those obtained from four previous studies on clearance levels for the reuse of materials and buildings of a nuclear power plant in the table (13). Note that the values listed from each of them are the most limiting from the various exposure scenarios considered. This comparison validates the results by showing that the DCGLs for all of the radionuclides, except Co-60, were less stringent than the generic risk-based ones established by the IAEA and in other studies, no matter whether considering industrial or residential use. These differences are mainly due to two reasons: (1) the exposure scenarios and parameters, and (2) the maximum doses considered. And so, these DCGLs are optimized values as they give the real minimal remediation levels required and hence
optimize the release process and the associated costs.

Table 13 A summary of the results of four studies on clearance levels for the reuse of materials and buildings of a nuclear power plant.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>IAEA (71)</th>
<th>IAEA (71)</th>
<th>CEC (72)</th>
<th>Haristoy et al. (73)</th>
<th>NUREG (74)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cs-137</td>
<td>4.00e-05</td>
<td>4.00e-03</td>
<td>5.00e-04</td>
<td>7.00e-04</td>
<td>8.00e-05</td>
</tr>
<tr>
<td>Cs-134</td>
<td>4.00e-04</td>
<td>4.00e-04</td>
<td>3.00e-04</td>
<td>2.00e-04</td>
<td>2.00e-05</td>
</tr>
<tr>
<td>Sr-90</td>
<td>1.00e-03</td>
<td>4.00e-03</td>
<td>2.00e-04</td>
<td>2.00e-02</td>
<td>1.00e-04</td>
</tr>
<tr>
<td>Ni-63</td>
<td>3.00e-01</td>
<td>1.00e+01</td>
<td>5.00e-02</td>
<td>3.00e+00</td>
<td>3.00e-02</td>
</tr>
<tr>
<td>Co-60</td>
<td>1.00e-05</td>
<td>1.00e-05</td>
<td>5.00e-05</td>
<td>2.00e-04</td>
<td>2.00e-05</td>
</tr>
<tr>
<td>Fe-55</td>
<td>9.00e-03</td>
<td>5.00e-01</td>
<td>4.00e+02</td>
<td>4.00e-01</td>
<td>8.00e-03</td>
</tr>
<tr>
<td>C-14</td>
<td>2.00e+00</td>
<td>1.00e-02</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>H-3</td>
<td>1.00e+02</td>
<td>3.00e-01</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Table (14) shows that DCGL values of the industrial scenarios, as was expected, are less restrictive than the ones calculated for the demolition and residential settings, which confirms that options for unrestricted use were appropriate for dose calculations. These values dictate the type and extent of cleanup, and their comparison allows to order the alternatives according to how realistic they are, and consequently, how much they cost. The residential scenario, being an unrestricted use with a low release level of 5.14e-07 Bq/m^2, is, due to that, the least realistic and most costly to achieve, in addition to it precluding any waste disposal areas as these must be located in restricted zones. Because containment rather than remediation after demolition would violate cleanup compliance, a DCGL for release had to be established, although higher than those for restricted use. That is mainly because the site will, in any case, enter a dormancy period that will allow radioactive decay, final cleanup, and then unrestricted release, and whose length depends on what’s intended for the site afterward.

Table 14 DCGL_gross results for the four scenarios.

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Scenario 1</th>
<th>Scenario 2.1</th>
<th>Scenario 2.2</th>
<th>Scenario 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>Demolition</td>
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<td>1.70e-05</td>
<td>1.68e-05</td>
<td>5.14e-07</td>
</tr>
<tr>
<td>Nuclear reuse</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Non-nuclear industrial reuse</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Residential reuse</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

5.2 AHP Results

In his book Beyond Decommissioning: The Reuse and Redevelopment of Nuclear Installations, Laraia [19] proposed ten evaluating criteria that are important for assessing the redevelopment potential.

1) ‘Decontamination cost’, which is the cost to reduce, remove, or neutralize radiological, chemical, and/or biological contamination to achieve the defined ‘acceptable’ level of risk exposure that is a prerequisite to reopening the site.

2) ‘Radiological safety’, which is the many controls implemented to maintain personnel and environment radiation doses below regulatory limits and as low as reasonably achievable, and to prevent the release of radioactive material into the environment.

3) ‘Project costs’, which is all costs and expenses a project will likely incur to be
completed.
4) ‘Economic value’, which is the commercial income generated from site reuse.
5) ‘Environmental impact’, by which is meant here only the adverse change to the environment resulting from the scenario’s activities, products, or services.
6) ‘Energy trends’, which is the combination of enabling trends and demand trends driving energy transformation, Sweden being the examined market here for the different energy sources.
7) ‘Nuclear energy policy’, which is all the country-specific and international policy tools concerning some or all aspects of nuclear energy.
8) ‘Expected users’, which is all the users or consumers for whom the scenario’s activities, products, or services are intended.
9) ‘Public opinion’, which is the average judgment or consensus of the individuals of the public concerned with the end state and reuse plan for O3 after its decommissioning.
10) ‘Time to open the site’, which is the time until the site becomes ready for its new purpose considering radiological safety.

In most decision-making, some of the considered criteria can be in the opposite direction to others, namely returns versus costs, and opportunities versus threats. Although this study does not derive negative numbers for priorities of costs and adverse impacts, here they are still referred to as negative priorities. It used the fundamental scale of positive absolute values of the analytic hierarchy process to rate the scenarios in terms of any criterion, and hence to estimate how much more advantages or disadvantages a scenario bears than the other ones. It derived the final priorities by multiplying each scenario priority with respect to a given criterion (table 16) by the criterion’s priority (table 15). The way to ordering the four reuse options was that the scenario with high negative priorities (fig.41) is disadvantageous and less likely to be chosen, and vice-versa; the opposite applied to the positive ones (fig.42).

Table 15 The derived priorities of the ten evaluating criteria.

<table>
<thead>
<tr>
<th></th>
<th>Decontamination cost</th>
<th>Health and Safety</th>
<th>Project costs</th>
<th>Economic value/benefit</th>
<th>Environmental impact</th>
<th>Energy trends</th>
<th>Policy</th>
<th>Expected users</th>
<th>Public opinion</th>
<th>Time to open the site</th>
<th>Priority</th>
</tr>
</thead>
<tbody>
<tr>
<td>Decontamination cost</td>
<td>1</td>
<td>1</td>
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<td>1</td>
<td>5</td>
<td>3</td>
<td>9</td>
<td>0.141</td>
<td></td>
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<tr>
<td>Radiological safety</td>
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<td>1</td>
<td>4</td>
<td>5</td>
<td>1</td>
<td>7</td>
<td>4</td>
<td>0.141</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Project costs</td>
<td>1/3</td>
<td>1/5</td>
<td>1/7</td>
<td>1/4</td>
<td>1/5</td>
<td>1/4</td>
<td>1/7</td>
<td>1/4</td>
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<tr>
<td>Economic value/benefit</td>
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<td>1/5</td>
<td>1/7</td>
<td>1/4</td>
<td>1/5</td>
<td>1/4</td>
<td>1/7</td>
<td>7</td>
<td>0.134</td>
<td></td>
<td></td>
</tr>
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<td>1/4</td>
<td>1/4</td>
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<td>1/5</td>
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<td>0.117</td>
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<td></td>
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<td>Energy trends</td>
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<td>1/5</td>
<td>1/5</td>
<td>1/4</td>
<td>1/7</td>
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<td></td>
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<td>1/7</td>
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<td>1/7</td>
<td>1/7</td>
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<td>1/7</td>
<td>1/7</td>
<td>1/7</td>
<td>1/7</td>
<td>1/5</td>
<td>1/6</td>
<td>1/3</td>
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</tr>
</tbody>
</table>
Table 16 The derived priorities of the four reuse scenarios, compared to each other with respect to the ten evaluating criteria, one at a time.

<table>
<thead>
<tr>
<th>Criteria</th>
<th>Demolition</th>
<th>Industrial nuclear</th>
<th>Industrial non-nuclear</th>
<th>Residential</th>
<th>Priority</th>
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<td>1/5</td>
<td>1/9</td>
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<td>1/7</td>
<td>1/9</td>
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<td>7</td>
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</tr>
<tr>
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<td>1/5</td>
<td>1/7</td>
<td>1</td>
<td>0.035</td>
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<tr>
<td>Time to open the site</td>
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<td>1/5</td>
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<td>0.117</td>
</tr>
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<td>1/7</td>
<td>1</td>
<td>0.033</td>
</tr>
</tbody>
</table>
6. Discussion

In anticipation of the decommissioning of the third unit of Oskarshamn NPP, this thesis performed radiological modeling that serves as conservative guidance for estimating the requirements of transforming the site into an acceptable condition, suitable for its envisaged future use. Using this information, with the MCDA results, this section identifies the scenario that is most optimal for the site, given opportunities, challenges, and unknowns in the four scenarios.

In decommissioning projects, preliminary remediation goals (PRGs) are used to identify initial cleanup goals that satisfy the site release criteria, and to weigh different reuse scenarios.

---

Figure 41 The derived negative priorities of the four reuse scenarios, compared to each other with respect to five evaluating criteria, one at a time.

<table>
<thead>
<tr>
<th>Criteria</th>
<th>Demolition</th>
<th>Industrial nuclear</th>
<th>Industrial non-nuclear</th>
<th>Residential</th>
</tr>
</thead>
<tbody>
<tr>
<td>Environmental impact</td>
<td>5.031</td>
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<td>0.3744</td>
</tr>
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<td>Policy</td>
<td>4.2108</td>
<td>0.3993</td>
<td>2.541</td>
<td>4.9489</td>
</tr>
<tr>
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<td>0.3993</td>
</tr>
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<td>Decontamination cost</td>
<td>0.52452</td>
<td>1.7625</td>
<td>4.4133</td>
<td>7.3884</td>
</tr>
</tbody>
</table>

Figure 42 The derived positive priorities of the four reuse scenarios, compared to each other with respect to four evaluating criteria, one at a time.

<table>
<thead>
<tr>
<th>Criteria</th>
<th>Demolition</th>
<th>Industrial nuclear</th>
<th>Industrial non-nuclear</th>
<th>Residential</th>
</tr>
</thead>
<tbody>
<tr>
<td>Time to reopen the site</td>
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<td>0.48</td>
<td>0.371</td>
<td>0.033</td>
</tr>
<tr>
<td>Public opinion</td>
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<td>0.4104</td>
<td>0.29</td>
<td>0.0945</td>
</tr>
<tr>
<td>Expected users</td>
<td>0.3648</td>
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<td>2.2116</td>
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</tr>
<tr>
<td>Energy trends</td>
<td>1.2546</td>
<td>3.0294</td>
<td>5.6202</td>
<td>0.2958</td>
</tr>
<tr>
<td>Economic value</td>
<td>0.4556</td>
<td>5.2528</td>
<td>5.2528</td>
<td>2.4254</td>
</tr>
</tbody>
</table>
objectives, free release being the ‘ideal’ scenario. It is after deciding for the site that PRGs are again used to establish the final cleanup levels. Learning of the latter’s importance aligns with the priority scores for both the decontamination cost and the radiological safety produced by the AHP, being the highest (0.141) of those for all the other criteria considered. At the stage of determining the final status of the site, residual radioactivity is expected to be below the DCGL value, and doses from likely scenarios exceed not 0.1 mSv/yr. Two levels of remediation, an expected and a high level, are adopted for restricted and unrestricted use respectively.

While two schools of thought, reuse versus demolition and new build, exist, this thesis argues that adaptive reuse is the next prime alternative to adopt for a decommissioned nuclear facility/site, and sets clear its priority over demolition and start-over developments anew. Although demolition is not the most optimal scenario, it should still be noted that, in most facilities to be decommissioned, cutting and demolition of concrete components or surface scarification to remove active or contaminated areas is needed. Full demolition requires large amounts of readily available money to be completed and should only be adopted when the building structure is unsuitable for the planned reuse, being too small, or too old, or having significant structural defects.

While demolition is assumed to require major remedial efforts and hence costs, its calculated DCGL\textsubscript{gross} = 6.35e-05 Bq/m\textsuperscript{2} determined as to the need for lesser cleanup than expected, when in fact the process of eliminating the environmental hazards is by itself environmentally harmful. The extent of demolition should be up to one meter below grade if all structures are bound to meet the site’s DCGL\textsubscript{gross}, and the excavation surface would be continuously surveyed to ensure that all nuclide-DCGLs are met. After the verification survey is completed, the excavation area would be backfilled with clean soil and material similar to the natural surrounding one, and then graded as necessary to restore it to a near initial appearance. Hereafter, accounting for the environmental cost would only reveal that the true cost of cleanup in the demolition scenario is by no means cheaper than that in the industrial ones, and whose DCGLs are of the same order of magnitude as that of demolition.

It is worthwhile to note that the removal of contamination sources does not necessarily render the site nonradioactive. But because demolition aims at unrestricted release, experience shows that the site must enter a dormancy period of up to 25 years to allow the licensee the time to meet the ‘no danger’ criterion, that is the risk of death to an individual does not exceed one in a million (10\textsuperscript{-6}) per year. In practice, this means removing all radioactivity even if negligible, but the question is whether this ‘over’ conservative approach is sustainable. The same shall apply to the residential scenario, which, among other reasons explained later, makes it the second least optimal option.

While many operators are required by regulations or have a corporate strategy or a financial incentive to minimize the amounts of decommissioning waste, demolition is the last scenario that can make it happen. The value for environmental impact is high in demolition because, in addition to airborne and waterborne emissions, excavation and backfilling,
the volumes of waste it produces are roughly equivalent to what is generated during the lifetime of the plant. [77] Estimated waste volumes from other demolition experiences of BWRs with similar main operation and design parameters are: [77]

- 12,460 m$^3$ of construction and demolition debris
- 2,832 m$^3$ of low specific activity
- 7,930 m$^3$ of class-A waste
- 88 m$^3$ of class-B waste
- 255 m$^3$ of class-C waste
- 40 m$^3$ of mixed-low level waste
- 680 m$^3$ of transuranic waste

As just shown, construction and demolition wastes represent a high volume of the waste produced and are assigned a "special waste" status, with a limit of 5.00e-04 mSv/yr for any residual radioactivity on site, which is far more restrictive than the 0.1 mSv/yr dose limit for a member of the public. Contaminated materials are decontaminated and free released for either reuse or recycling, where practical and economically feasible. Investment in recycling and reuse pays back within a short time, which explains why demolition’s priority is low for decontamination cost and high for economic value, adding to that the avoided cost of ongoing site surveillance and maintenance.

Where recycling is impossible, materials are disposed of as nonrecyclables in an inactive landfill. Admittedly, waste transport and disposal in an offsite repository is a great benefit safety-wise. Buildings that no longer have ‘significant’ value are demolished, something the market determines. Such an assessment may not be the most accurate as it usually discounts externalities and non-monetary values. However, to demolish a building in an assessed fair/good condition, only to transport the remains to a landfill that could be a few kilometers away in some cases, but a few thousand in others, and for which the cost can be prohibitively high, is not the best plan when other alternatives, that promise fewer costs and more benefits, exist.

A point made by Laraia [19] is that, although redevelopment is not necessarily less expensive than new construction, its costs typically lie within the same order of magnitude, making it a more sustainable option. Critical to choosing it over other alternatives is the rising cost of new developments associated with the increasing energy costs. And the cost is of particular relevance in decommissioning projects because usually, it is not the sole problem of the licensee; its social aspect conditions the way decisions for future use and their financial implications must be achieved.

So the question is: is costly complete demolition the best use of limited resources and best practice of social equity? From budgetary, environmental, and socioeconomic perspectives, the answer is no. Redevelopment saves much of these resources by avoiding waste production due to demolition, reducing the raw materials consumed in new construction, and reducing development pressure on open space by reusing land that has already been developed, especially true in the case of industrial reuse. To redevelop rather than
to abandon a decommissioned site is also a decision that provides a positive perception of decommissioning and increases workers’ motives and capacities as they know their work will be for the tangible benefits of people.

It is true that a building is also demolished when its life expectancy is evaluated to be shorter than after remodeling or rebuilding, which seems more logical and economically sensible as it would be unnecessarily costly and time-wasting to redevelop a retired facility to a desirable, stable state than to demolish and rebuild. However, this study argues that this is a short-sighted viewpoint. Looking further in the long run at the importance of generating economic benefits and hence minimizing ongoing costs, backed by the added social and environmental rewards, there won’t be fighting choices: redevelopment is the way to go.

Safety-wise, concerns of potential later radiological exposure are eliminated in demolition and not in the industrial scenarios. The solution to that is progressive delicensing and in-parts redevelopment, starting with the so-called ‘buffer zones’ which are typically external to operations and contaminated areas. This way, the redevelopment process is initiated while contamination is allowed to decay, and funds for the decommissioning of the contaminated parts are secured.

The lower (expected) remediation level required, and the reduced ‘footprint’ that is the area to be decommissioned, hence the reduced waste volumes, make the industrial scenarios indeed more attractive. Despite that, worker health and safety are maintained at all times. The relaxed release requirements for industrial development also encourages investors to purchase the site at a low price while the current license holder(s) cannot afford its cleanup. And after they complete the cleanup (at a low cost), investors often sell the place at a higher price or redevelop it for profit.

While research has shown that experience with restricted release has been difficult for reasons becoming less prominent with innovation over time, such as failure of administrative and legal controls, complexity in dose-response modeling, and power given to municipalities to impose special terms and conditions including permits, some legislations and government directives have established restricted instead of unrestricted release as the preferred option. The norm has changed over the years, with unrestricted release becoming less favored, mainly due to its technical unfeasibility and the significant, often unrealistic, costs of bringing the site of a nuclear reactor back to where it could be used as a playground. Despite that, buffer zones, at least, can still be redeveloped as (1) parks since they qualified for active development of largely intact natural habitats over many decades and (2) as houses with the current need for additional real estate assets and space and more sustainable communities. [19]

The main reason why the unrestricted release was ideal and why the term for it is ‘greenfield’ is that it was regarded as the only environmentally appropriate end-state of decommissioning, and whose goal is to restore a nuclear site into its ‘pristine’ condition. However, and given the change of local communities during an NPP’s lifetime, attention turns to public opinion, and it would only be fair to apply a greenfield strategy if the new
locals accept it. With that said, locals mostly prefer a greenfield status premised on social-psychological bases for environmental and radioactivity concerns over brownfields. Yet, paradoxically, the same concerns prevent people from living or going to a free released nuclear site. Environmentally speaking, but from a perspective other than radioactivity, things have changed with adaptive reuse bringing many environmental benefits such as land recycling and improving the building’s energy efficiency, material strength, and environmental performance. [78]

Now whether industrial redevelopment for nuclear or non-nuclear use highly depends on the most recent trends in energy followed by cost and environmental impact. What’s remarkable is that, despite the differences in parameters and nuclide DCGLs, the gross release levels of both scenarios were very close, and thus, remediation-wise, the same will apply whichever option is adopted.

While evidence shows that nuclear is better to expand with recycling and reuse of decommissioned sites wherever possible, a question to start with is: is the nuclear share in an increase? Figure (43) shows the installed and projected power capacity in the world between 2000 and 2040 in the new policies scenario, one that looks at the impact of existing government policies and commitments on energy demand, supplies, and investments.

![Figure 43 Total installed power capacity in the world under the New Policies Scenario, 2000-2040. (IEA, 2020)](image)

Nuclear (as can be seen in fig.43) seems to be making a slight additional contribution, with its share of total generation increasing from 10.15% in 2019 to just 11% in 2040. That is due to a lot of factors, including the fact that new nuclear power plants have proved to be expensive in recent years, often involving delays and cost overruns, and that their eventual decommissioning is viewed and dealt with as a terminal phase. This slight increase in the world is to be paralleled by a major projected decline in Sweden as by the early 2040s, all its reactors will have been withdrawn from service, and its business community being uninterested in building new ones. Under its Energy Policy Agreement, Sweden aims
for a 100% renewable electricity supply by 2040. Something that deserves a separate mention is that this is a goal, not a deadline that prohibits nuclear power, nor does it require a political decision to shut down nuclear reactors.

Based on all of that, the most optimal use of the site at release is industrial non-nuclear. This study reiterates that to be able to integrate these arguments and reach this conclusion, it is necessary to update the process of license termination and site release where the endpoint should go beyond site release and consider post-decommissioning reuse. (fig.44)

Examples from operating experience in the reuse of decommissioned sites include:

- A biodiesel plant at Greifswald site in Germany.
- Companies at the Oak Ridge K-25 site in the USA including a recycle center; a tool and die shop for the automotive industry; a medical isotope production facility; a company of specialized trucking services; and a factory of infrared heating systems.
- A fully equipped mechanical workshop at the RB-2 reactor in Italy.

*Figure 44 Conclusion of the decision-making process of this study, and which shows the added steps to the ‘conventional’ nuclear site release process flow diagram. (AlAli, 2020)*
7. Conclusions and future work

Radiological modeling serves as conservative guidance for estimating the requirements of transforming the site into an acceptable condition, suitable for its envisaged future use. From budgetary, environmental, and socioeconomic perspectives, costly complete demolition is not the best use of limited resources and best practice of social equity. Looking further in the long run at the importance of generating economic benefits and hence minimizing ongoing costs, backed by the added social and environmental rewards, there won’t be fighting choices: redevelopment is the way to go. Evidence shows that nuclear is better to expand with recycling and reuse of decommissioned sites wherever possible, but a question to start with is: is the nuclear share in an increase? And while the answer is no, the scenario that is most optimal for the site in 2045 (or 2055) is industrial reuse for non-nuclear purposes.

Recommendations for future work include but are not limited to:

- A sensitivity analysis to evaluate the extent to which the numbers that represent the judgments of experts and the evaluating criteria in the AHP matrix have on the findings and conclusions.
- A feasibility study to assess the practicality of the proposed scenario.
- An additional study to identify, design, and develop a specific non-nuclear use for Oskarshamn unit 3.
- Future studies to evaluate redevelopment options for the remaining operational reactors in Ringhals and Forsmark in Sweden.
- Further development of this thesis’s work, and which could prove beneficial to the literature.
8. References


