行政院原子能委員會

委託研究計畫 期末報告

噴砂除污技術於核電廠除役之應用探討

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中文摘要

目前國際之核電大國已有充足之除役拆廠經驗，其經驗與規範可供我國核電廠除役之參考。其中，放射性除污技術應用的適用性評估對於除役施工期間相當的重要。過去核電廠在尚未除役前所使用到的除污技術，一般來說只是針對核設施進行檢修之用，即使技術上可能與核電廠除役時所需類似，但其與除役實際進行的規模相較之下相差甚遠。由是之故，美國、加拿大、日本、歐盟等國，為了除役作業的大規模除污做了相當多的研究。根據既有的資料來看，許多的除污技術有著共同的缺點：操作溫度高、除污時間長、產生的廢料多等，也因此除了傳統的物理、化學除污之外，近年來研發的除污技術朝向縮短除污時間、減少廢料量和提高除污效果的方向發展。本計畫主要是針對噴砂除污方式進行探討，瑞典在噴砂除污方面的經驗是以濕式噴砂為主，其認為濕式噴砂相較於其他除污方式有較好的效果，且廢棄物也易於處理與貯存。

Abstract

At present, many countries with nuclear industries have sufficient decommissioning experiences, in which the experiences and regulations can be for reference. The assessment of radioactive decontamination technology for application is quite important during decommissioning construction. The decontamination technology used in nuclear power plant under normal condition is to overhaul for some specific nuclear facilities. The scale of decontamination technology for normal condition is quite different from that for decommissioning, even the technology is almost the same. Therefore, many countries, such as America, Canada, Japan, EU, have made several studies on decontamination technology. According to the information about the decontamination, many techniques have the common defects, like high operating temperature, long-term decontamination, more waste generated. In addition to traditional physical and chemical decontamination technologies, new types of decontamination technology have been developed recently, such as plasma, microwave, and introduce them on decommissioning. This project focuses on the blasting decontamination approaches. The experience of the blasting approach for Sweden is mostly to use wet blasting technology. The result showed that wet blasting has better effect than other contamination technologies, and the waste is easy to handle and keep in storage.

壹、研究背景與目的

1.1 前言

於 2011 年 3 月 11 日日本福島核電廠發生嚴重核子事故，在電視與網路媒體的強力報導下，使得國內與全世界人類對於核能電廠安全運轉產生了相當大的疑慮，倒核聲浪更是與日俱增。雖然台灣使用之核能機組與日本同樣源自於美國之設計，且在基準設計上與日本核電廠相較之下更為保守。因此，加強核能安全之相關研究，以福島事件與其他過往的核災事故作為借鏡，精進我國核安管制能力，確保核安管制品質，以增進社會大眾對核能安全之信心。

現階段我國政府所採行的能源政策主要是以確保核能安全、打造綠能低碳環境並逐步邁向非核家園作為總體能源發展方向。在此能源政策為主軸下，國內的核能電廠未來勢必將傾向不再延役，將面臨運轉屆齡四十年而必須採取除役的方式來終止運轉執照。此外，已封存的核四廠也已確定不會運轉，且其內 1744 根燃料束將會送回美國原廠。因此，國內三座核電廠的除役計畫勢必將會逐步展開。縱使核電廠在除役階段所產生核能事故的風險遠低於核電廠正常運轉，然而人為操作、除汙方式等使用方式不當亦會對人類與生態帶來汙染影響。因此，在除役階段正確的處理方式與工作態度是相當重要的一環。對於核電廠之除役作業，除了經營者需要針對除役工作之執行、管理、技術研發等進行妥適的規劃之外，管制單位也應事先精進除役作業相關的專業知識與管制基礎，並培植必要的管制人力，以因應未來核電廠面臨除役時的審查及管制工作。故在除役過程中將會面臨許多核能安全相關問題與可能遭受意外事故之挑戰，這些挑戰將是可以被預期的、透過分析評估，可提出圓滿解決問題之答案。

台灣電力公司核一廠一號機組，將於民國 107 年底運轉到期，國內相關研究

單位也持續針對核電廠除役安全評估、除污技術、場址評估等進行相關的研究與評估。然而，實際上國際在核電廠除役技術上所累積的專業技術與實務經驗，與我國在除役的作法上仍有些不同；更何況，國內過去並未曾實際進行過核電廠的除役作業。因此，掌握各國核子反應器設施之除役法規、實務經驗及關鍵安全問題與評估技術，從中獲取重要的經驗與技術資源，以協助國內核電廠各項除役作業計畫，使除役計畫得以順利執行，並有效率地完成除役工作。

國際上為了對核反應爐設施的除役能提供一致的方法，除了吸取先前已除役或正在進行除役工作的核能電廠所獲得的經驗外，國際上認可關於安全評估、除污方式或除役相關之重要報告中必須有核能電廠的除役導則。所謂的除役是指允許對一座核能設施解除部分或全部監管所採取的行政管理行動和技術行動。這些行動包括放射性材料、廢物、組件和建築物的去污、拆除。執行這些行動是為了有計劃逐步減少放射危害，為了確保除役操作期間的安全，這些行動都是根據事先的計畫和評定採取的。核能電廠的除役活動時間可以從幾年至幾十年，其中會考慮到放射性衰變的因素。故除役可以在核反應器關閉後以連續操作方式進行，亦可以在一段時間內以間斷操作的方式進行，此即為分階段除役。然而，對任何除役工作與核能設施繼續運行之間的相互影響所可能引起的安全問題，則將需要根據當時的具體情況做進一步的處理。

國際上核電廠除役實際應用的情況會因國情不同而有所改變，許多作法應用在我國除役的電廠上並不能照本宣科，必須依國內所需作進一步的修正改進，以符合國內之情況。

1.2 研究目的

核能電廠在進行除役規劃時，工作人員、大眾、環境生態的保護是管理組織上最重要的環節，如何針對除役期間除污作業準則、程序、技術、作業安全、輻射防護等進行審慎的規劃，以確保執行時能降低廢棄物的污染，且將是國內首次進行核能電廠除役工作時之重要課題。此外，核電廠除污方式是以降低廢棄物污染為主要準則，但減少工作人員輻射曝露與廢棄物的有效減量亦是必須同時考量為目的。

除污的目的是為了降低輻射曝露、回收利用舊設備和材料、減少需要特別處置的廢棄物體積，使場地和設施恢復到不受限制使用的狀況，也須將鬆散易擴散的放射性污染物固定在原處，做好永久封存和最終處置的準備。除污技術的原理是以物理、化學、電化學、生物分解、熔融等方法為主，過程中牽涉到的事物包含了：待除污物件、除污工具、被分離出來的污染物質、除污後的剩餘物件基材、額外加入的除污輔助物質、用來清潔的材料及清潔後產生的混和物質等等，這些東西有的是一般廢棄物，有的是放射性廢棄物，依使用技術不同而有不同的特性與數量。

本計畫將針對固體廢棄物處理程序方法中的噴砂除污技術進行探討，針對國際上現有的技術與應用進行資料彙整，並進行噴砂除污技術之瞭解，將有助於增加國內未來在執行相關作業時的基礎；同時可強化審核單位自身之能量，以應日後相關審查工作所需。

1.3 研究工作

本計畫將歸納整理噴砂除污技術的原理和技術基礎，評估其除污效果及其適用範圍，所獲得之成果可協助未來管制單位在制定系統除污安全規範上之參考依據。以下為本計畫之工作項目與甘特圖。

1. 蒐集彙整核電廠除役之噴砂除污技術相關資訊與經驗

蒐集國際上除役相關法規等資訊，藉以瞭解各國核能電廠除役法規對於噴砂除污技術之作法與應用，並藉此了解國內除役法規與各國除役法規之差異。本研究主要將參考 NEA 於 2011 年時所發表關於放射性廢棄物的報告進行噴砂除污技術相關資料的彙整；而 IAEA 於 2001 年所發表的報告亦有詳細的噴砂除污技術資料，亦為本研究之參考。此外，國際上在研磨劑材料種類、各種研磨除污方式的比較等，如表 1 與表 2 所示為整理之資料。本研究將進行蒐集彙整與研析，瞭解國際中對於噴砂除污技術之應用經驗，以作為國內核電設施除役計畫可供參考之審查規範與研訂。

2. 國外除役電廠實際經驗案例分析

國外除役電廠之案例已有相當多之經驗與累積，雖國內核能法規是以美國為基準，但在此計畫除了將蒐集美國相關核電廠除役實際案例外，於其他核電大國於除役之經驗案例亦採取廣泛蒐集彙整供作參考。對於除役時所應用之噴砂除污技術，將針對可供參考之實例進行完整及深入之收集。

表 1-1 研磨劑材料種類與特性

材料種類	耐磨性	污染性	Mohs'硬度	比重	形狀	消耗度
矽砂	大	大	7.0	2.5	不定形	大
鋼	大	大	7.5	7.0	球形	小
氧化鋁	中	中	9.0	3.8	不定形	小
玻璃珠	小	小	5-6	2.5	球形	小

表 1-2 各種研磨除污方法之比較

	乾冰除污	grit blasting	高壓水	弗立昂	冰	清潔劑/化學清洗
表面破壞	否	是	可能	否	是	是(酸)
去除鬆散污染物	是	是	大部份	大部份	大部份	是
去除固著污染物	是	是	否	一些	一些	否 是(酸)
產生廢水	否	是(採用水噴射時)	是	否	是	是
產生固體廢棄物	HEPA 過濾器	預敷過濾器、樹脂及 grit	預敷過濾器及脂	筒式過濾器	預敷過濾器及脂	預敷過濾器及脂
產生混合廢棄物	否	可能	可能	是	可能	可能(酸)

年月 工作項目	1	2	3	4	5	6	7	8	9	10	11	12	備註
廣泛蒐集及研讀核電廠除役之噴砂除污技術應用相關之法規及文獻					*	*	*	*	*				
國外電廠實際經驗之案例分析，同時充分掌握與了解國外已除役或即將除役之核能電廠的現況							*	*	*	*			
針對除役核電廠的噴砂除污技術進行研讀與分析，提出歸納整理								*	*	*	*		
提出歸納整理之研究報告與相關之建議事項									*	*	*	*	
工作進度估計百分比（累積數）	%	%	%	%	%	%	%	%	%	%	%	100%	

圖 1-1 研究進度甘特圖

第二章 國際除役相關之文獻

美國、德國與日本等核電使用大國皆具有相當豐富的除役相關經驗，在其除役計畫和除役結果報告中均存有十分寶貴之資料與實作紀錄，可供我國核電廠進行除役作業時之參考。雖然各國電廠之設計與環境條件不同，除役計畫亦須依各國國情做調整，其並無法一體適用，且需針對各電廠進行審慎評估與檢討，才能擬定最適用於各電廠之除役計畫。我國核電廠係以美規為主，各類的規範與法規大多皆以美國核電廠為導向。美國 NRC 對核電廠除役需求訂定相關法規與法規指引及出版 NUREG 報告，如 RG 1.184 說明了除役方法、除役階段及時程等規定； NUREG-1700 提供除役審查導則、NUREG-1757 分成三大冊內容，並提供美國核管會的除役程序、方法、技術、財務規劃等，這些對於國內核電廠除役提供了相當重要的參考依據。不過其他國家的除役經驗與相關重要資訊對於尚未有過除役經驗的我國而言，皆是重要的參考依據。

本計畫主要是參考國際原子能總署(International Atomic Energy Agency, IAEA)、經濟合作與發展組織核能署(Nuclear Energy Agency of Organisation for Economic Co-operation and Development, OECD/NEA)以及美國核管會(Unite States Nuclear Regulation Commission, NRC)等之文獻報告。其中，IAEA 技術報告中之 No. 399 “Organization and Management for Decommissioning of Large Nuclear Facilities (2000)” 闡述除役核能設施的各項步驟和基礎方向。NEA 於 1999 年出版的”Decontamination Techniques Used in Decommissioning Activities” 以及 OECD/NEA RWM-R2011 的技術報告”Decontamination and Dismantling of Radioactive Concrete Structures”對於除污方式與技術也都有詳盡的說明。

2.1 美國的除役管制

美國核能管制的聯邦法規是將權限授予各州的聯邦政府及機關，以一般規定的形式，通過立法(聯邦規定)施行法令。核物質處理與利用之相關的聯邦法規，即核能法(AEA)是將許認可權限授予核能管制委員會(NRC)，其遵照聯邦規定法令之第 10 項(10 CFR：NRC 規定)，經由各項管制的實施，施行法令。這些 NRC 規定並透過 NRC 所出版的法規指引(RG)、NUREG 等管制導則，以及業界出版的國家工業規格等，進行了內容上的補全。

這些規定中確立了所有與除役相關的規定，包括：除役許認可、輻射防護、副產物物質許認可、低放射性廢棄物的包裝、運送及處置等。訂定核能設施除役一般要件的相關規定之「除役措施規定(Decommissioning Rule)」，其 Part 30、40、50、70 及 72 係以適用於取得 NRC 許認可的核設施，為其制定目的。依除役規定之要求，制定適當的除役計畫，並經由實施該除役計畫終止執照，使得廠址解除管制，得以無限制使用。

管制機關不限定反應器執照經營者只能選擇 DECON(立即拆除)或 SAFSTOR(延遲拆除)，亦可合併運用 SAFSTOR 及 DECON 兩種模式的組合。反應器執照經營者因計畫目的，亦可將核能設施安全貯存一段時間後，再進行拆除大型機器設備，如：蒸汽產生器、加壓器及反應器內部組件等。接著將機組設備安全貯存約 30 年時間，然後再完成整個除汙及除役程序。然而，因現行法規規定除役須在停止運轉後的 60 年內完成，故只有在必須保護大眾健康與安全的情況下，才會考量追加除役時間。

在地下處置場興建完成到可供使用狀態之前，可先將用過核燃料繼續貯存於用過燃料池或乾式貯存護箱內。另外，在規定方面要求要設計成 ISFSI(用過核燃料獨立貯存設施)，以供除役之用。而 ISFSI 永久停止使用前，也應做好事前準備工作，使放射性廢棄物及受污染設備減至最少，以便輕易除去放射性廢棄物及污染物質，進行結構物及受污染機器設備的除汙工作。

2.2 英國的除役管制

英國核能除役局(NDA)為了制定適用於全英國有關核設施除役及淨化方面的重要導則，依據 2004 年能源法案在 2005 年 4 月成立，是不隸屬於保健安全管理局轄下的公家機關。NDA 的職責是以公正且資訊透明的方式，適切地考量除役帶來的社會及經濟層面的影響，並考慮到英國民生用核能遺產設施的安全防護與成本效益下，制定國際間適用的除役計畫。NDA 進行公開協議，並取得英國政府的認可,在 2006 年 4 月時公佈了從 2006 年到 2011 年為主的初次導則。根據 2005 年的能源法案，有必要每 5 年重新檢視一次導則。且 NDA 負起這些廠址所有除役相關的資金調度及導則之責任，於執行除役措施業務相關之職權內，與廠址經營者及廠址運轉執照公司(SLC)締結合約。NDA 不僅需承擔這些廠址的商轉及廢棄物管理活動之責任，同時也擔負著廠址最終除役的責任。這部分的財源,一方面來自政府的供應,一方面是由 NDA 廠址的商業活動所得支付。NDA 將大部分的經費列在除役及除汙方面，並以此部分的費用為優先，進行風險管理的同時，也承擔著降低風險的責任。

2.3 法國的除役管制

法國在 1950 年至 1990 年間以來，陸續興建了許多的 INB，但其中大多數已停止運轉或進入解體階段，包括核能發電用反應器、研究用反應器、研究所、燃料再處理廠、廢棄物管理設施等，已於 2007 年時停止運轉或進行解體中。因此，與此相關之拆除作業的安全及輻射防護也漸漸成為了法國核能安全委員會(ASN)的重要課題。ASN 聽從 IAEA 的建議，主要是考量可避免帶給未來世代於拆除作業之技術性及財務上的負擔，故除役方式選擇了 DECON 方案。現今法國當前的所有主要運轉機關，也保證對目前預定解體的設施，將採取立即拆除方案。

而除役電廠所有的拆除作業皆達到法國核能安全委員會(ASN)認可的最終預期狀態時，設施即可宣告完成除役，並依照設施的最終停止運轉及拆除相關

之許認可政令所記載手續，得以從 INB 的名單中排除。申請除役的結束，須檢附含有土壤分析、殘留設施之建設的說明，特別是關於解體後廠址的預期狀態等內容的說明書。

在運轉機關所提出的許可申請的附件中，必須敘述從永久停止運轉開始至欲達成預定最終目標為止的整體工作計畫。關於各階段的手續，對於由設施所引起之潛在性危險的性質及範圍，以及因應處理方法之提案，也須提供詳細的資訊。永久停止運轉及解體的階段，可在最初運轉許可的管制架構下，一直執行到準備停機為止。這段準備期間是特別為了消除部分或全部的輻射源，以及為了準備拆除作業(配置廠房、準備現場作業、訓練團隊等)而設置的。

2.4 德國的除役管制

德國的除役措施，包含了核能電廠從最終停止運轉、轉讓期、拆除，以及核能電廠或廠址解除核能管制前的所有措施。德國對於核能設施除役的許可程序，是以核能法(AtG)、及根據 AtG 的法規及一般性行政規定作為法規上的依據，其中包含了除役之許可基本要件。該要件是針對所有被許可的設備，規定必須取得停機後，該設備或組件拆除、安全封存或解體方面的許可。在此同時，也應考慮以最先進之科學技術加以執行，此為應被遵守的原則。核能設施除役之許可程序，是根據 AtG 中的「設施許可手續相關法規 AtVfV（Atomrechtliche Verfahrensverordnung）」來規範。該法規中，關於除役的管制，特別包含了第三方的參與及環境影響評估(EIA)的相關規定。

核能設施除役主要是經由移除燃料束、最終移除超出解除管制基準(Clearance Level)的殘留放射性核種，以及透過解除核能管制，持續地減少核能電廠的放射性核種的總量(Inventory)。並且，當設施處於冷卻狀態，壓力也被除去時，與運轉階段不同，可說如放射性物質總量會擴散那樣的潛在性能量已幾乎不存在。大致而言，隨著拆除作業的進行，將同時持續地降低危險性。在以法令為主的管制架構中，藉由除役相關之具體管制及勸告、適用既有的管制架

構及配合降低可能風險，以及在辦理許可手續或監督手續的過程時，經由廢除監督規則及要件，證明其危險性確有持續降低。

2.5 德國 EWN/Greifswald 電廠除役[9]

德國EWN電廠除役因計劃規模過於龐大，於除役之初即擬定大組件拆除，經過中期貯存輻射劑量衰減後才開始進行切割處理。德國EWN/Greifswald 電廠拆除策略如下所述：

1. 以大組件拆除，貯存於廠內中期貯存設施。
2. 大組件貯存等待輻射劑量衰減後才進行切割處理。
3. 用過燃料貯存於廠內中期貯存設施。
4. 現場拆除需要降低劑量率，才使用現場除污。

由於該廠最後決定將先以大型組件拆除為主，在等待劑量率衰減後切割處理之策略，估計縮短除役時間約3 年。此外除了大型組件直接暫貯於ISN，其餘拆除廢棄物於Warm Workshop 進行切割檢整及除污，符合外釋限值者則進入外釋程序，否則仍運至ISN 暫貯。所使用的除污設備包括高壓水除污、噴砂除污、電化學除污及化學除污。以下簡述所使用的除污設備：

<u>高壓水除污</u>

除污包封 7.0 m 長 × 4.0 m 寬 × 4.5 m 高，配備2,000 bar 高壓水柱設備以執行除污。附屬設備包括1 公噸吊車，旋轉架，排氣及過濾系統，水處理系統。1人全副著裝於除污包封內操作除污設備，另1 人於包封外安全監視。

噴砂除污

除污包封 8 m2 × 2.5 m 高，配備10 bar 之手持噴砂設備以執行除污。附屬設備包括旋轉架，排氣及過濾系統，噴砂研磨料回收處理系統。1 人全副著裝於除污包封內操作除污設備，另1 人於包封外安全監視。

化學除污及電化學除污

磷酸除污槽 5 m^3 1 座，2.5 m^3 2 座。電化學除污槽2 m^3 2 座，2000A 及1000A整流設備各一套。除污劑不再生處理。

污染熱切割包封

切割包封7.0 m 長 × 4.0 m 寬 x 4.5 m 高，配備1 公噸吊車及熱切割設備，附屬設備有排氣及過濾系統。

其餘設備包括剪床、各式帶鋸切割機、及電纜剝皮機。

2.6 比利時電廠除役

自 1989 年以來，比利時核能研究中心(SCK•CEN)一直在拆除其壓水式反應器 BR3（Belgian Reactor N°3）。在實際拆除兩組內部構件期間，獲得了高放射性元件以遙控拆除(Remote dismantling)方式的許多經驗後，BR3 團隊完成了其反應器壓力容器(RPV)的切割。在 RPV 拆除的可行性階段期間，決定拆卸後的元件從爐穴移出之後，便在電廠內的燃料更換池內進行切割，並使用銑刀和帶鋸機將 RPV 切割成數段。這些機械技術已顯示出它們進行此種操作的能力。在進行分割之前，包覆於 RPV 外層的隔熱層會以遙控方式來移除和處理。BR3 除役作業(Decommissioning activities)也包含了拆除受污染的環路和設備，也執行了一些不同除污方法的開發，如噴砂(或噴砂)、化學去污(使用鈰的氧化還原過程)。

第三章 研究方法

3.1 除役各階段作業規劃

核一廠除役各階段工作時程規劃，主要分成四個階段，停機過渡階段、除役拆廠階段、廠址最終狀態偵測階段及廠址復原階段，如圖 3-1 所示。其中停機過渡階段約 8 年、除役拆廠階段約 12 年、廠址最終狀態偵測階段約 3 年，以及廠址復原階段約 2 年，共計 25 年。整個除役過程包含停止運轉後的用過核子燃料移除、系統除污、安全貯存、設施與建物的解體、廠址復原及釋出等步驟，其中更包含除役廢棄物管理、除役過程之善後與環境管理等重要步驟。核電廠設備、結構、組件的拆除與除污工程是整個除役過程的核心重要步驟，工作內容與範圍相當繁瑣。且經切割拆除後的設備所要面臨的首要問題即是除污方法的選用與執行，以及所產生之放射性廢料之後續處理與善後復原工作。除污作業在除役過程中是相當重要的一環，應針對不同的除污物件選擇不同且適合的除污技術與方式，並使最終的目的朝向有利於環境保護、人員健康、節省開支、避免資源浪費。此除污的工作階段如圖 3-2 所示。當拆除與除污作業完成後，則會進行廠址環境輻射偵測，以及除役最後之階段-廠址復原階段。

放射性除污的基本概念是利用清洗、加熱、機械、化學作用或其他方式來除去核設施或相關設備表面上的放射性污染物，其目的是為了降低輻射曝露、回收利用舊設備和材料、減少需要特別處置的廢棄物體積，使場地和設施恢復到不受限制使用的狀況。近幾年國際上許多核電廠面臨除役作業的需求，因此對於放射性除污技術的需求逐漸提升。過去核電廠於正常運轉時的除污技術通常只是針對核設施的檢修為目的，雖然檢修所應用到的除污作業在基本技術上可能與除役相似，但在規模上卻相差甚遠。因此，許多國家為了除役作業的大規模除污方面做了非常多的研究。不過，就目前來看，這些除污相關技術都有

著共同的缺點：操作溫度高、除污時間長、產生的廢料多等。因此，近年來世界各國都想盡辦法解決這些問題，朝向縮短除污時間、減少廢料量和提高除污效果的方向發展。經過多年的除污技術研究，化學、機械、材料冶金、生物分解等多種類的除污技術不斷被開發出來，於是除污技術的研發有了很大的進展。

本計畫的噴砂除污工作在整個除役過程上係屬除役拆廠階段，此階段主要目標為開始進行全面放射性污染之系統、設備、組件之拆除，拆除物件包含核一廠聯合結構廠房、汽機廠房及其他污染建築物內之系統、設備、組件等，以及廠房內受污染之混凝土結構物表面污染之刨除。配合拆解與拆除作業，所產生的低放射性廢棄物需要進一步處理者，如量測、處理/判別、整理/分類、減容、組件除污、包裝運送等，尤其是除污部分為本計畫所要探討的重點之一。並將依據低放射性廢棄物類別與輻射特性進行分類處理，處理完之低放射性廢棄物使用已獲許可之貯存容器包裝，並存放於已獲運轉執照之貯存設施。

3.2 除污技術敘述

除污的目的是為了降低輻射曝露、回收利用舊設備和材料、減少需要特別處置的廢棄物體積，使場地和設施恢復到不受限制使用的狀況，也須將鬆散易擴散的放射性污染物固定在原處，做好永久封存和最終處置的準備。一般對除污的定義是：透過物理、化學或其他方法移除核設施組件、系統、結構物內、外表面存在的放射性物質，要達成此項目的必須先理解放射性核種如何與物體表面結合，才能選擇適合的除污方法[10]。

一般來說，放射性核種和物體表面的結合形式有：

1. 以分子間作用力結合的非固定性污染。此種污染容易去除，因核種和物體表面並無發生化學反應，結合較為鬆散之故。

2. 以化學吸附或離子交換形式結合的弱固定性污染。此種污染因二者之間產生較為緊密的化學結合，通常伴隨一定的滲入情形，除污難度略高。
3. 放射性核種擴散滲入基材內部或基材材料內微量元素被中子活化產生放射性核種。這種情形的污染牽涉到內部的大量滲入，固定性很高，因此很難除污。

依據放射性核種和物體表面的結合形式不同，採用的除污方法也不同。表3-1為不同的放射性污染除污方法說明。以核電廠除役時之管路、設備及組件拆除後之除污作業而言，拆除作業進行前之管路、設備及組件內部的除污，目的是減少輻射物質，降低人員執行拆除工作所接受之劑量。而拆除作業完成後所進行之除污，目的是將表面污染之管路、設備及組件進行清理，使物料達到有條件外釋或無條件外釋的標準。常見的除污方法可分為化學、電化學及機械方式：

1. 化學方式

   化學方式是將物料浸泡在溶液中，使用稀釋或高濃度的各種化學試劑來處理污染表面，以溶解金屬基底或其表面的污染層。化學除污技術主要可分成三大類：A.螯合作用；B.酸性或鹼性溶解及C.氧化還原反應。

2. 電化學方式

   電化學處理金屬廢棄物的表面污染，是將鋼料放入電解槽中並作為陽極，通以直流電而產生陽極的溶解，此程序又可稱為電解拋光(Electropolishing)；其優點是處理快速且可靠度高，若能將電解液作有效的回收再利用，可使二次廢棄物的產量最小化。電化學除污技術僅能用於具導電性的表面，技術之限制為電解槽的體積、待處理表面的幾何形狀及清潔度的要求。電化學除污技術無法應用在具有複雜幾何形狀之表

面。

3. 機械方式

大口徑管路或設備外殼以機械方式進行表面除污。通常是採用磨料噴砂除污技術，可建立大型帳棚防止污染擴散，並於除污完成後進行輻射偵檢。此外，無論是以化學或機械方式完成大面積除污後，對於部分殘留的熱點，將採用研磨、刨刮或銑削等機械方式加以移除。如圖3-3為利用高壓噴水方式進行除污清洗，圖3-4為利用以砂為研磨劑之噴砂除污方式。

本計畫所探討的噴砂除污方法即為物理法，常見的物理除污方法有沖洗、吸塵、機械擦拭、高壓水/蒸氣噴射、磨料噴射表層剝離、超音波除污等方法。主要又可分為三大類：機械除污法(又細分為表面淨化與表面移除)、物理表面清洗技術、熱清洗表面方法。

表 3-1 放射性污染的除污方法[10]

除污方法	敘述
物理法	利用沖洗、吸塵、機械擦拭、高壓水/蒸氣噴射、模料噴射表層剝離、超音波除污等方法，對汙染物進行物理效應的處理
化學法	利用化學溶劑清洗污染的區域和設施，以達到除污的目的
電化學法	利用電解反應，使造成表面污染的放射性核種進入電解槽中，以達到除污的目的
熔融法	將低度污染的金屬廢料經熔煉處理後，使大部分的放射性何種進入爐渣或尾氣中，少部分殘留於鑄錠內呈現均勻分布
生物法	利用微生物對放射性污染物的生物分解或吞食作用，除去這些物質，以達到除污的目的

在金屬材料的除污方面，目前對於金屬材料的除污多半採用二氧化碳丸(乾冰粒)的方法加以處理。1970 年代，美國在應用乾冰的過程中發現了固體二氧化碳的除污性能，一直到了1980 年代，乾冰清洗技術才實際被應用出來。在乾冰清洗系統中，液態二氧化碳經由乾冰製備系統備固化成乾冰方塊，然後再經研磨而成規格一致的乾冰粒，接著使用壓縮空氣加速乾冰粒通過噴嘴而噴射至物體表面，當乾冰粒被噴射至物體表面時，乾冰粒破裂所產生的動能滲入基體材料，並使其散開，側向噴出的碎片，去掉了基體表面上的污染物，使污染物掉落地面。由於乾冰碎片立刻昇華產生上升力，從而加速污染物的去除。過程中揮發的二氧化碳不具毒性，可以直排大氣，產生的廢物少，與噴砂、噴冰等的去污方法相比有明顯的優勢。對清洗塑膠、陶瓷、複合材料和不銹鋼也都有效。二氧化碳雖無毒性，但仍屬窒息性氣體，在密閉環境中能使人窒息死亡，工程上須提供良好的自然通風環境，才能避免工安意外。目前此技術的瓶頸在於乾冰粒的製備，並無法有效率地大量製備，此技術也因此受到限制。

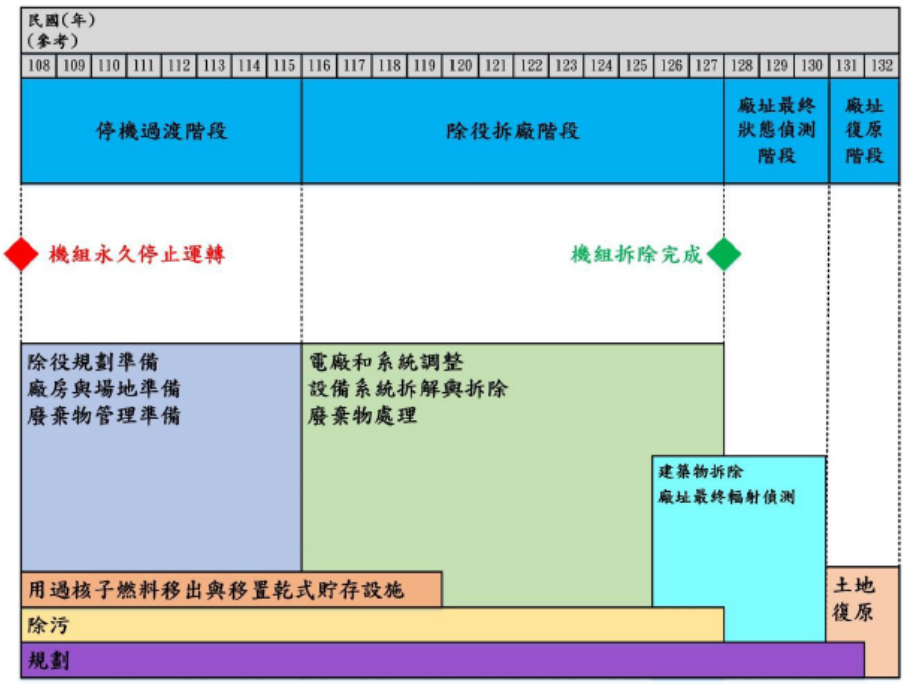

圖 3-1 核一廠除役架構時程圖[6]

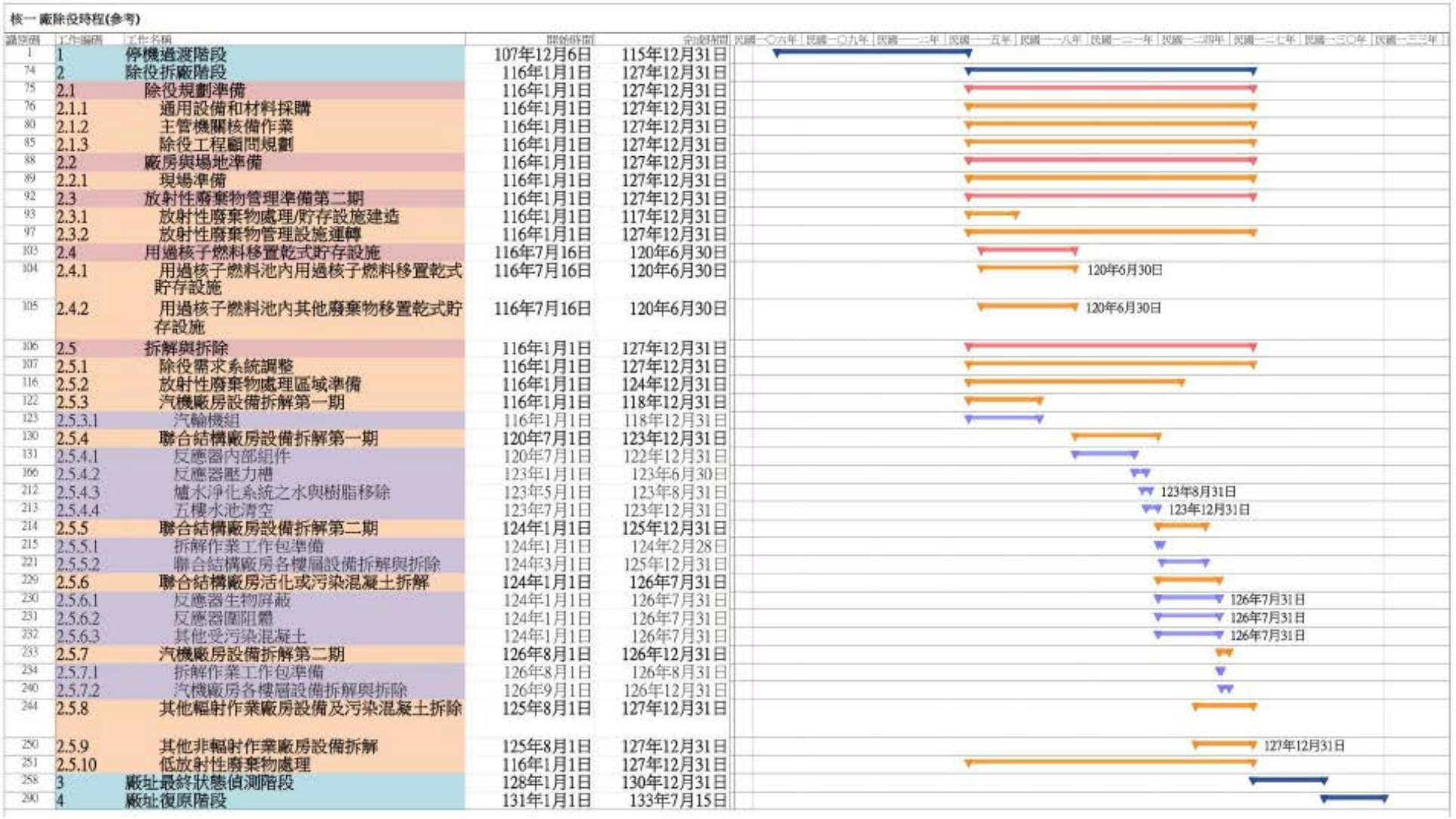

圖 3-2 除役拆廠階段作業排程[6]

圖 3-3 高壓水噴射之除汙作業方式[15]

圖 3-4 研磨料為砂之噴砂除污作業方式[15]

3.3 噴砂除污技術

國內相關之主管機關應在核電廠實際進行除役前，須先建立完整之相關審查技術規範，以提升審查效率及嚴謹度，並同時增進國內核安相關審查主管機關之審查能力、專業知識與相關重要技術之瞭解。

拆除作業進行前之管路、設備及組件內部的除污，目的是減少輻射物質，降低人員執行拆除工作所接受之劑量。而拆除作業完成後所進行之除污，目的是將表面污染之管路、設備及組件進行清理，使物料達到有條件外釋或無條件外釋的標準，常見的除污方法可分為化學、電化學及機械方式。針對多孔性表面，化學或電化學除污技術並不適合，而可利用機械除污技術進行除污。核一廠除役時規劃在聯合結構廠房、汽機廠房建立此類技術的使用設備，針對拆除或切割後之組件，進行表面除污程序。

在機械方式部分，大口徑管路或設備外殼以機械方式進行表面除污。通常是採用磨料噴砂除污技術，可建立大型帳棚防止污染擴散，並於除污完成後進行輻射偵檢。此外，無論是以化學或機械方式完成大面積除污後，對於部分殘留的熱點，將採用研磨、刨刮或銑削等機械方式加以移除。而此機械方式除污在除役作業規畫中，列入二次廢棄物管理規劃，主要是著重於除污作業可能衍生的二次廢棄物及減廢措施。噴砂研磨技術有乾式或濕式兩種方式，使用乾式噴砂研磨時，須在工作區域加裝粉塵控制系統及真空過濾系統，以降低粉塵污染狀況。使用濕式噴砂研磨時，會產生大量的廢水、磨料及研磨後的碎屑粉塵，過程中應採取適當的減廢措施，如磨料的回收及廢水(包含處理或未處理)的再循環等。上述組件除污作業所產生之廢棄噴砂可盛裝於 55 加侖桶或固化處理。圖 3-5 與圖 3-6 為其噴砂研磨的應用。以下將介紹這兩種方式[11]。

濕式噴砂沖洗系統是一封閉環路與液體研磨除污技術，系統利用水、研磨劑與壓縮空氣結合，一般應用係於獨立與氣密之不銹鋼外殼內，無懸浮污染危險，因其具有絕對過濾器之獨立的通風系統，可維持機殼內呈負壓狀態。放射性廢

棄物可利用機械分離方法(如旋風分離器/離心分離、篩分)自清洗介質中分開，水經過濾後可循環，無需可溶性或有害化學藥劑。

。除污作業現場應設置排氣設備及高效率過濾器，並在除污作業區內維持負壓，避免空浮污染之情況發生。

濕式研磨清洗可應用於金屬表面可擦拭與固著污染之去除，例如結構鋼、臺架、組件、手工具及機械零件等之除污。此設備亦可應用於無需移除金屬之密閉零件如汽機葉片或閥除污，改變空氣壓力與研磨劑可用來去除重型腐蝕與塗料。

圖 3-5 於地板應用之研磨噴砂之商業化設施[11]

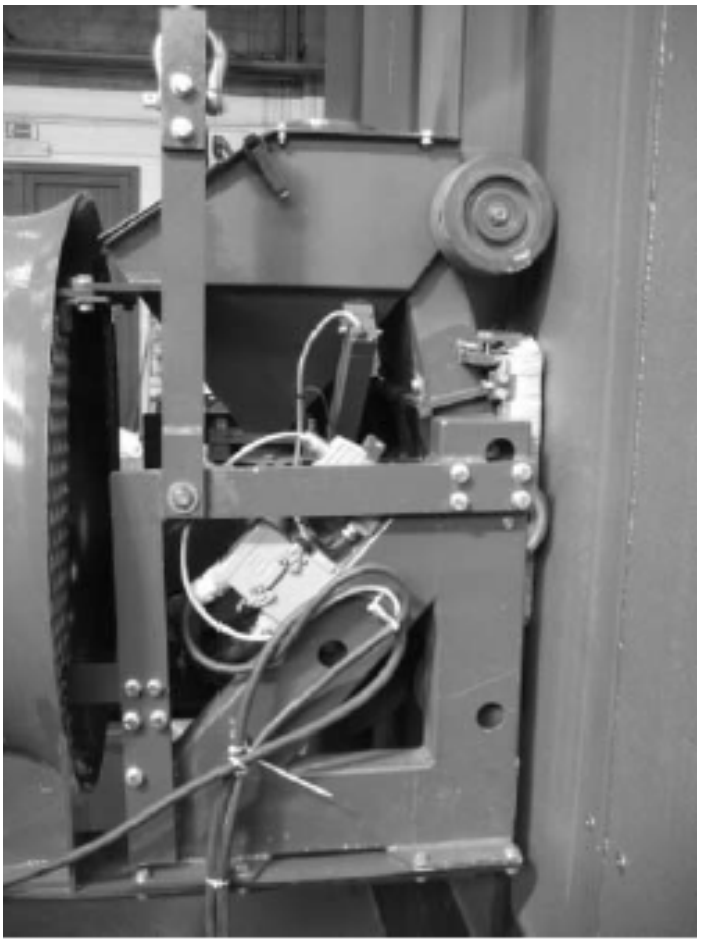

圖 3-6 垂直是研磨噴砂應用[11]

乾式噴砂沖洗一般稱為噴砂或研磨噴射，此技術係利用研磨物質懸浮於一介質，然後射出至待除污物件表面，均勻地去除表面污染。一般以壓縮空氣或 blasting turbine 來攜載研磨劑。所移除表面物質與研磨劑收集，利用旋風分離器將研磨劑與移除表面物質分離，研磨劑循環再利用。除了會被研磨劑打碎的物質如玻璃等之外，其他物質之表面污染均可應用乾式噴砂沖洗進行除污，但應避免應於鋁與鎂，因有粉塵爆炸之危險。

乾式研磨噴砂技術，通常以壓縮空氣或噴射渦輪設備帶動研磨材料，對表

面進行快速噴射研磨料顆粒的除污效果顯著，產生清潔、除鏽、除毛邊的作用，將金屬、塑膠或石造表面移除污染物質。噴砂技術可應用在平面開放式的表面，包含地板及牆面，也可應用在具有複雜表面的設備或組件。根據核一廠待除污之金屬廢棄物，可選擇各種適合的研磨材料，包含：(a)礦物類，如磁鐵礦或砂；(b)不鏽鋼顆粒或氧化鋁；(c)玻璃珠、玻璃融塊、碳化矽及陶瓷材料；(d)塑膠粒子及(e)二氧化碳乾冰。研磨材料具有不同型體(圓形、不規則狀、棱角狀)、硬度及密度，可根據待除污之金屬表面選擇適合之研磨材料。例如金屬表面的塗漆可選用圓形或低硬度之磨料，而單純金屬表面可選用棱角狀或高硬度之磨料，以達到除污效果

上述研磨材料並不包含矽材，以避免作業人員引發矽肺病。另一方面，核一廠規劃採用高壓水除污技術，以高壓噴射水強力沖洗污染表面。可溶性的污染物會溶解，而鬆散未固著的顆粒則藉著水而被帶走。高壓水技術可以應用在金屬表面，亦可針對混凝土、磚材、磁磚表面進行除污，無論是多孔性或無孔表面都有良好的效果。不過此技術不建議使用在木材、纖維或相似的材質上。上述機械除污作業規劃程序如圖3-7所示。

研磨可在濕式或乾式條件下應用，在乾式條件下應用時，須有粉塵控制對策以控制粉塵和/或空浮污染，在作業區使用真空過濾系統可減少粉塵和/或空浮污染問題。在濕式條件下應用，會產生大量廢棄物包括廢水、研磨劑及移除碎屑，過程中應採取適當的減廢措施，如磨料的回收及廢水(包含處理或未處理)的再循環等。上述組件除污作業所產生之廢棄噴砂可盛裝於 55 加侖桶或固化處理。這些廢棄物必須妥善處理和/或處置。研磨劑再循環及水處理後再利用可大幅減少二次廢棄物產生量。

核一廠現有之乾式除污噴砂設備，設置於汽機廠房(標高 73.83 ft)處，相對

位置如圖 3-8 所示，利用磨料進行物件之表面除污，其砂料重複回收使用。依據核一廠的空間，可將噴砂研磨除污區域設於聯合結構廠房，如有需要則可搭配廠內原有廢液及廢氣處理系統，進行二次廢棄物之處理。機械除污區域必要時應裝設獨立式(Self-Contained)排氣設備及 HEPA，並在作業區內維持負壓，避免空浮污染的情況發生。具放射性的廢棄物以機械式的方式分離(如旋風分離、離心或過篩)收集，而廢水可以過濾並循環利用。

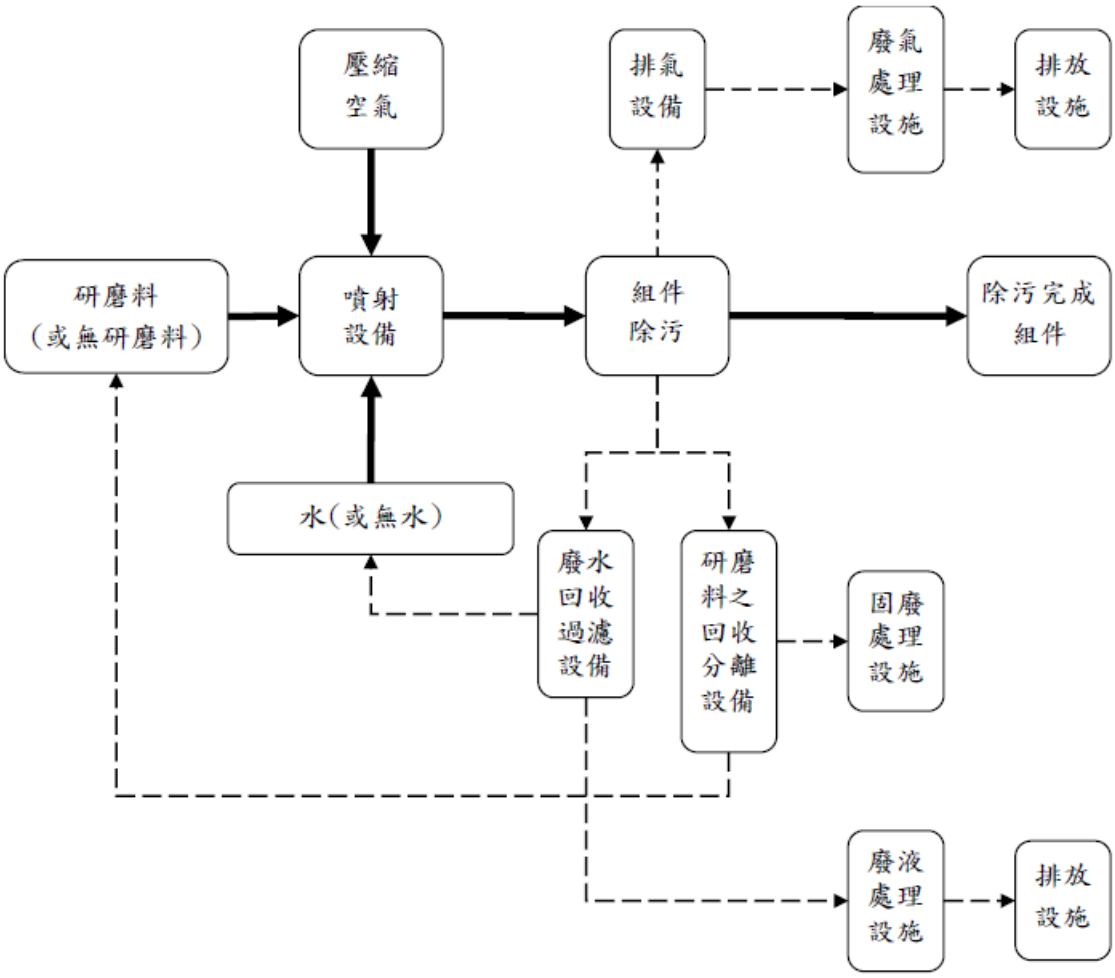

圖 3-7 核一廠金屬類組件之機械除污作業流程示意圖 [7]

表 3-2 為在除污工作期間,使用乾和濕研磨料的噴砂系統進行除污工作所產生之二次廢棄物的一些結果，其中耗水量包括在每一次去污循環後，以清水清淨各組件所耗費的水量。

表 3-2 濕式噴砂除污對於金屬除污之結果[16]

	Dry abrasive blasting	**Wet abrasive blasting**
Efficiency	Very high	Lower
Grit consumption	55 g/kg metal	109 g/kg metal
Secondary-waste production		
– Intervention clothing	5.3%	8.2
– Grit waste	5.5%	10.9%
– Water consumption	–	6.9 1/kg metal
Decontamination rate		
– Plates	57.4 kg/h	48.0 kg/h
	2.8 m²/h	2.3 m²/h
– Profiles	127.7 kg/h	106.8 kg/h
	1.8 m²/h	1.3 m²/h
Grit cost	0.5 ECU/kg	2.25 ECU/kg

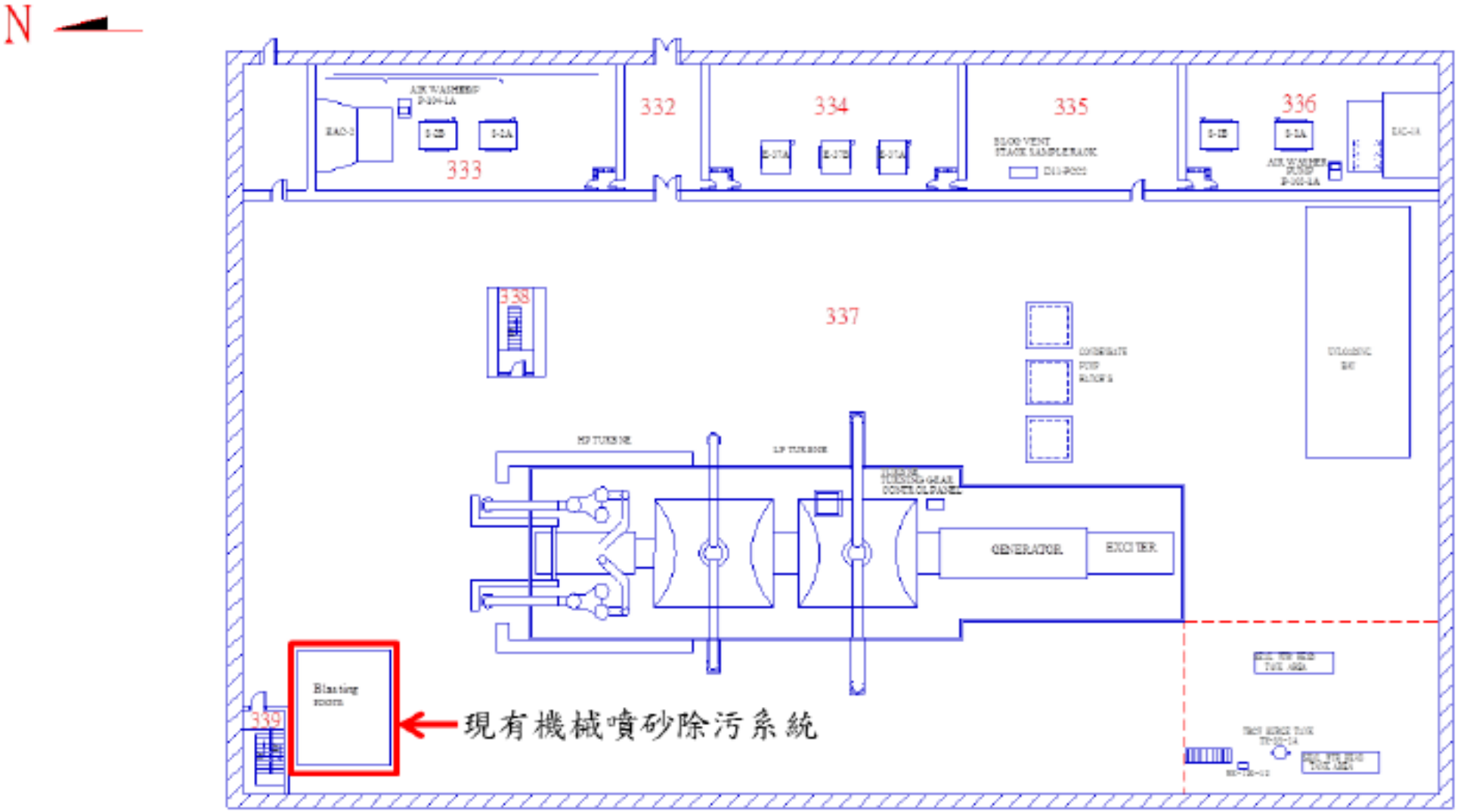

圖 3-8 核一廠現有機械噴砂除污系統位置圖 [8]

組件之機械除污技術可分為表面清潔及表面移除兩大類，如同化學及電化學組件除污，組件之機械除污設備應設置在核一廠除役作業規劃之廢棄物處理區域中，以方便集中管理。組件機械除污時會產生懸浮粉塵污染，因此，須裝設防止污染擴散之設備及人員安全保護設備。採用濕式噴砂研磨是屬於封閉迴路系統之技術，結合水、壓縮空氣及研磨料，在負壓狀態櫃體中進行除污動作，並須裝設含有高效率過濾器之空氣通風系統。放射性廢棄物以旋風/離心分離器進行篩選移除，而廢水則加以過濾並循環再利用。乾式噴砂研磨利用高壓空氣帶動研磨料對待除污組件進行衝擊研磨，除去組件表面材質達到除污效果。乾式噴砂研磨不適用於會被研磨料擊碎的材質，如玻璃或樹脂玻璃(Plexiglas)等，以免破碎造成危險。乾式噴砂研磨避免運用在鋁及鎂材質表面以降低導致粉塵爆炸的風險。採用乾式噴砂研磨技術之前，須先將可燃性污染物移除(如紙類、木質類等易燃物先予撕除或磨除)，以避免除污時污染物產生受熱燃燒或爆炸的風險。另外，乾式噴砂研磨除污過程中會產生靜電，須將待除污組件或設施裝設接地設備。

以下為噴砂系統的應用。

研磨噴砂是一種多功能技術，它可以構建特定的安裝以應對特定的廢棄物，例如混凝土屏蔽塊或容器。在 Belgoprocess 其用於處理混凝土容器的設施廢棄物，如圖 3-9 為使用在混凝土容器的除污方式。這種安裝可以使用內部金屬對大型混凝土容器進行淨化塗層，為了最大限度地降低額外污染的風險，所建立的安裝是用於混凝土材料和鋼襯的除污。這套設施是安裝在通風的空間內，其中包括三個隔間並在內部和外部進行噴砂。產生的灰塵可藉由空氣和過濾器來清除其污染物。在此操作之後，所有容器的表面確認是否仍有污染。圖 3-10 為乾式噴砂為小型混凝土作業之安裝。

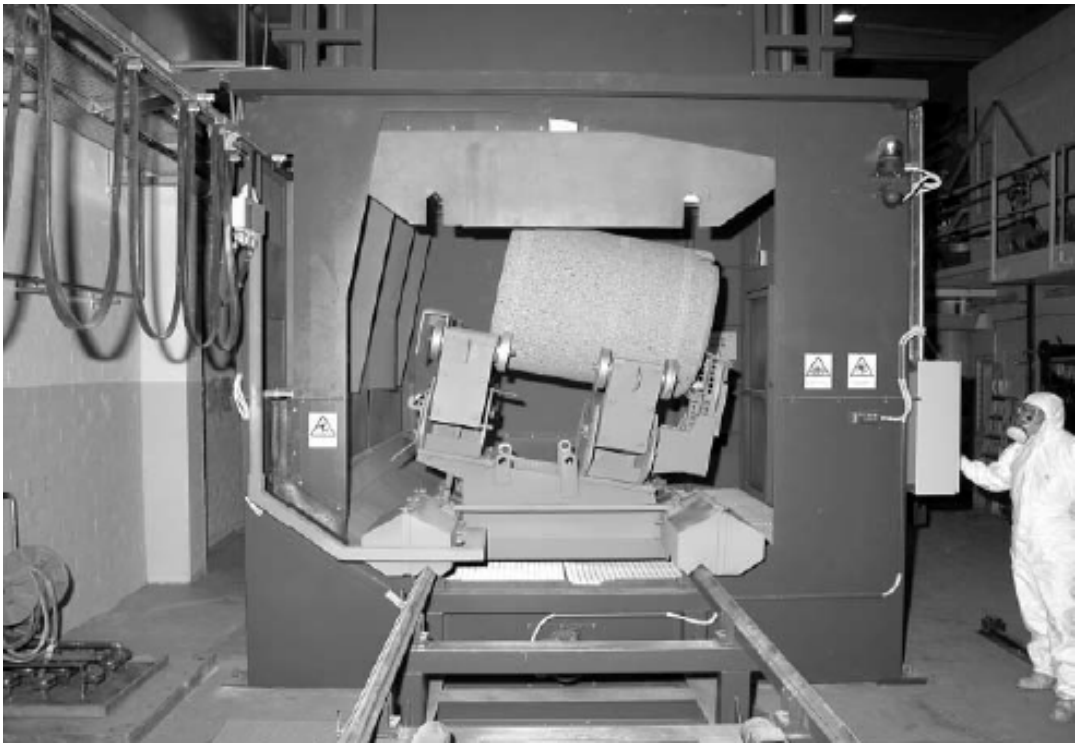

圖 3-9 使用在混凝土容器的除污設備[11]

圖 3-10 使用乾式噴砂於小型混凝土容器[11]

3.4 除役除污階段之相關準則

核電廠除役的目標、階段、時程，拆除系統設施所使用之設備、方法、除污技術應用及安全作業程序，必須要有清楚的階段步驟，以利於明確釐清除役相關的作業程序，達到廠區與人員輻射安全。以下幾點條列除役的相關準則[]：

1. 明確說明除役各階段時程。
2. 明確說明除役完成時間，以符合法規定。
3. 以Gantt或PERT圖明確說明各階段拆除程序及時程。
4. 明確說明各結構、系統、設備拆除之安全作業程序。
5. 明確說明各結構、系統、設備拆除程序之輻射防護設計。
6. 廢棄物種類及數量估算。
7. 人員劑量估算。

其中，除污計畫是本計畫之重點項目，所需注意的準則包含除污範圍、除污標準、除污作業準則、除污方法。除污作業包括除污之物件如拆除前之系統除污、場地除污、工具及拆除器具除污、金屬廢棄物除污、待減容廢棄物除污、防護衣物除污、現場廢棄物收集處理等，其使用的除污方式如機械除污、化學除污、電化學除污。除污作業必須考量限制二次廢料量的產生，其廢棄物包括放射性物質與非放射性物質建立資料庫。以下幾點條列除役的相關準則：

根據放射性活度調查及初步評估結果擬定除污計畫。

1. 明確說明除污範圍及除污標準。
2. 明確說明除污方法及方法流程圖。
3. 估算除役期間放射性廢液產量。
4. 明確說明限制二次廢料量規劃。
5. 明確說明放射性廢氣、廢液處理程序。

6. 明確說明放射性廢氣、廢液超出排放標準之處理程序。
7. 根據放射性活度評估結果、除污作業、拆除方法估算廢棄物數量。
8. 明確說明減廢措施。
9. 根據系統、設備、組件拆除順序及動線，規劃廢棄物處理、運送、貯存方法。
10. 說明廢棄物最終處置。
11. 說明設備、材料或土地之回收與再利用。

第四章 國際使用噴砂除污技術探討

4.1 金屬元件以噴砂方式除污(比利時)

在 1991 年 Belgoprocess 開始了工業級規模的拆除 Eurochemic 再處理廠後處理設施。為了持續控制廢棄物處理、中間及最終處置的成本，測試項目顯示放射性材料的除污達成無條件清潔基準(Unconditional clearance level)有很好的成果。基於此結果可建立一測試程序來驗證不同除污技術的可行性。Eurochemic 再處理廠的營運期間從 1966 年起至 1974 年止，專門處理來自核電廠和研究用反應器的核燃料。Eurochemic 再處理廠的主體建築是一個大型的混凝土結構，由一個混凝土體積 12,500m^3 以及 1500 噸的金屬組件所組成。經過除污之後，超過一半的混凝土以及近 70%的金屬，已被無條件從廠區釋出處置。在 2008 年裡，主體結構的重要組成部分已經被拆除，並且整個除役工作將於 2012 年完成[17]。圖 4-1~4-3 為除役時的情形。

圖 4-1 核設施除役之前的情形[17]

圖 4-2 拆除工作進行中的情形[17]

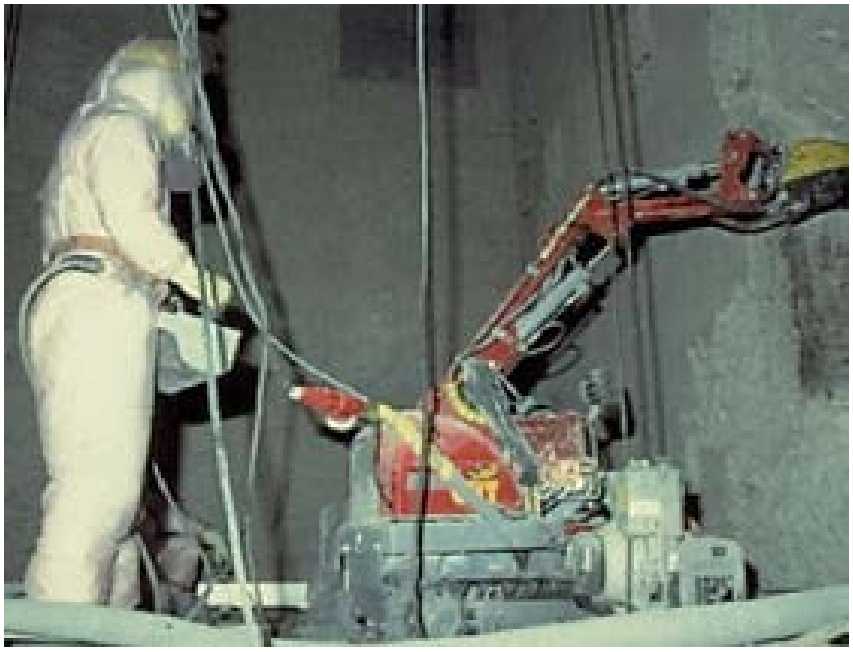

圖 4-3 進行混凝土除污工作時的情形[17]

以一個 18 噸、受汙染且生鏽的碳鋼除污為例[18]，其正是以噴砂系統方式來進行評估。且其提及，不同的化學技術測試並無法得到材料可接受的測試結果。然而，在實驗室規模的樣品上，藉由小型的噴砂系統的方法，可以得到不

錯的成效。根據這些實驗室等級的測試結果，Belgoprocess 開始了一驗證計劃來看 18 噸的碳鋼藉由乾式與濕式噴砂的成果。

另一方面，在這個驗證計劃中，藉由噴砂除污系統的成本與正常廢棄物處理相比，根據相同的材料進行了詳細的研究。保守估計在較為保守方式來看成本的比較，在此成本比較中，乾式和濕式研磨系統已納入在成本以內。表 4-1 與 4-2 分別為 9 噸碳鋼利用乾式或濕式噴砂系統的除污方式的成本比較表。

表 4-1　9 噸碳鋼利用乾式噴砂系統的除污方式的成本比較表[18]

Task	Decontamination by dry abrasive blasting	Supercompaction and disposal
Capital costs	1,279,-	
Installation costs	358,-	
Cutting and transport	619,-	619,-
Reducing and packaging		1,669,-
Decontamination	841,-	
Measurements	54,-	
Waste treatment	817,-	3,963,-
Intermediate disposal	35,-	175,-
Final disposal	200,-	1,000,-
TOTAL COSTS	4,203,-	7,426,-

表 4-2　9 噸碳鋼利用濕式噴砂系統的除污方式的成本比較表[18]

Task	Decontamination by wet abrasive blasting	Supercompaction and disposal
Capital costs	3,795,-	
Installation costs	307,-	
Cutting and transport	619,-	619,-
Reducing and packaging		1,669,-
Decontamination	840,-	
Measurements	54,-	
Waste treatment	640,-	3,963,-
Intermediate disposal	28,-	175,-
Final disposal	160,-	1,000,-
TOTAL COSTS	6,443,-	7,426,-

基於成本評估後，乾式與濕式噴砂系統都納入於除役的實施方法中。此外，其也評估了 309 噸的鋼也是採用研磨料噴砂系統進行除污，並也進行了成本評估，如下表 4-3 與表 4-4 所示。

表 4-3　309 噸碳鋼利用乾式噴砂系統的除污方式的成本比較表[18]

Task	Decontamination by dry abrasive blasting	Supercompaction and disposal
Capital costs	2,867,-	
Installation costs	10,251,-	
Reducing and packaging	32,106,-	64,212,-
Decontamination	10,629,-	
Measurements	6,464,-	
Waste treatment	13,855,-	135,850,-
Intermediate disposal	616,-	5,936,-
Final disposal	3,520,-	33,920,-
TOTAL COSTS	80,308,-	239,918,-

表 4-4　309 噸碳鋼利用濕式噴砂系統的除污方式的成本比較表[18]

Task	Decontamination by wet abrasive blasting	Supercompaction and disposal
Capital costs	40,000,-	
Installation costs	10,658,-	
Reducing and packaging	32,106,-	64,212,-
Decontamination	22,169,-	
Measurements	6,464,-	
Waste treatment	10,467,-	135,850,-
Intermediate disposal	448,-	5,936,-
Final disposal	2,560,-	33,920,-
TOTAL COSTS	124,872,-	239,918,-

4.2 濕式噴砂於瑞典的經驗

Johan[20]的研究報告式探討濕式噴砂在核電廠的研究，其中提到 Fagerstrom Industrikonsult AB and their division DECO 系統供應濕式噴砂艙作為使用。在濕式噴砂方面，其是用一 150~250m 之珍珠狀的玻璃與水作為介質來進行。這份報告並說明在兩個核電廠與貯存設施進行噴砂，其皆為沸水式反應器核電廠。在 Barseback 核電廠方面，其以兩部 BWR 機組，每一部皆為 615MW，分別於 1975 與 1977 年開始運轉。而其在 1999 年開始停止運轉，並於 1999 年開始進行除役。Oskarshamns 核電廠有三部機組(O1、O2 和 O3)，其分別為 445MWe、605MWe 和 1160MWe。反應器 O1 為瑞典第一部商轉反應器且在 1972 年開始營運。而反應器 O2 與 O3 分別於 1974 與 1985 年開始運轉。在除役方面，有一除污專門的部門有兩套噴砂艙供給這三座反應器來使用。

在瑞典有四座核電廠，其分別為 Barseback、Oskarshamn、 Ringhals 和 Forsmark。這些電廠的除污處理方面有一專門的部門來負責，而濕式噴砂為其中之一的除污方法。Barseback 和位在 Oskarshamn 的除污部門都有相似的噴砂艙，其是由 Fagerstrom Deco 系統所供給，如圖 4-5 所示。此噴砂艙內部空間為 5500mm x 3400mm x 2500mm，裡面可放置最重達 2000kg 的元件在支撐樑上。雙通道的門位於艙體短邊的其中一側，且與頂部的艙口一起可允許吊重機進行運輸工作。艙體本身為一個單位，載體泵浦(Media pump)、載體容器(Media tank)、分離容器(Separation tank)與淨空泵浦(emptying pump)將組合成一組為另一個單位，而過濾處理系統也是一個單獨的單位。

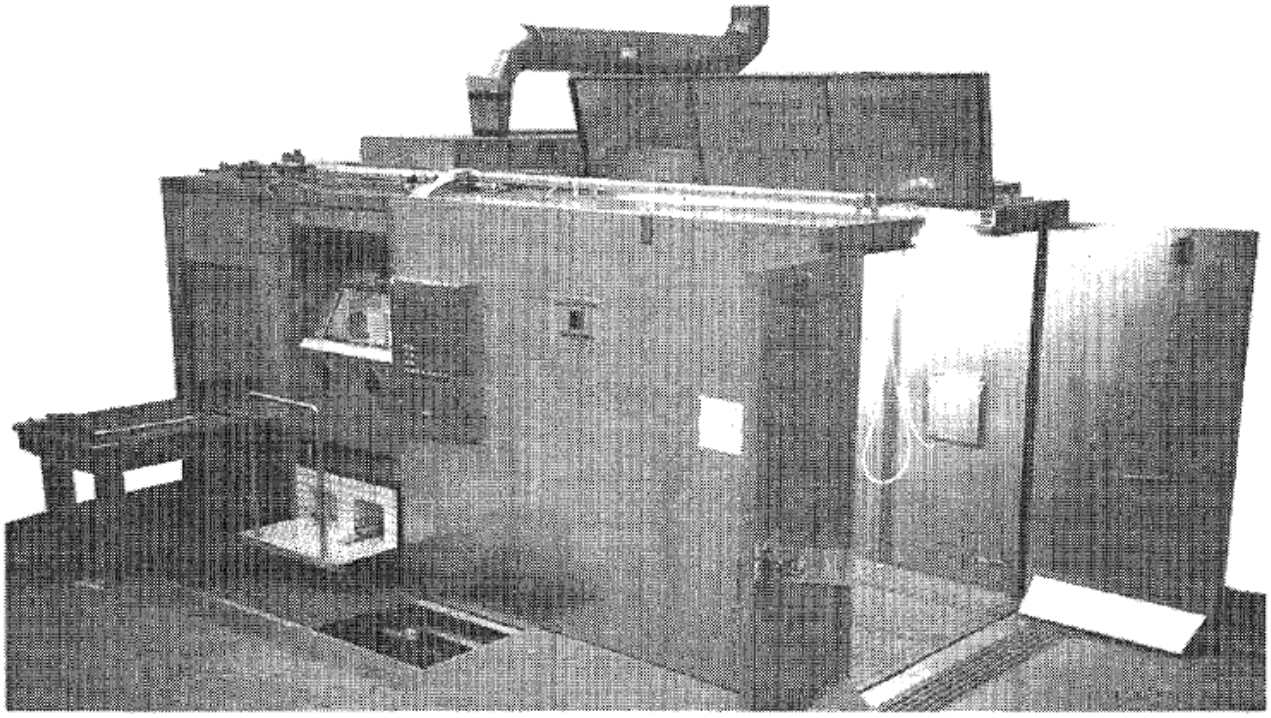

圖 4-5 位於 Barseback 的除污部門噴砂艙[20]

為了符合核能工業的需求，濕式噴砂除污系統是由冷軋不銹鋼 304 型表面 2B 級高拋光所製成。載體容器和管路系統是由耐酸不鏽鋼 316 型所製成，且某些元件如載體泵浦和閥件具有橡膠內表面。所有外部節點都有防水功能，且屋頂是以焊接方式緊密接合以便進行外部清洗。此外，為了盡量減少噪音，噴砂艙內以隔音材料來處理。濕式噴砂系統設計是同時以三個噴嘴來使用。在柵欄下面噴砂艙較低的部分是採用錐形設計的，並將載體送回到載體容器。所有的錐體都配備噴嘴沖洗系統以進行清潔，且濕式噴砂除污系統有兩個外部操作員的位置，其是採用輻射防護（1280kg）和 76-82 mm lead 玻璃窗（相當於 24.9mm lead）。

4.2.1 噴砂材料

使用的噴砂材料是玻璃，它是以珍珠的形狀來使用。其經事實證明玻璃珍珠在較硬的表面上具有拋光效果，幾乎不會使表面退化，如同採用砂子研磨材料的效果。用於噴砂的玻璃珍珠可以用不同的等級購買。但最適用於在硬質金

屬上噴離氧化層的是 150-250 μm。玻璃珍珠會隨著時間的降低其等級，但這並不會導致其表面品質降低。如果使用較大的等級，可能導致受噴的金屬表面產生硬化的可能。這可能不是我們所想要的結果，因此當不會造成任何可量測的表面硬化時，將會使用 150-250μm 的尺寸。從使用濕式噴砂設備的經驗是它是一種當使用正確尺寸的玻璃珍珠時，其可以去除氧化層而不損害元件。操作員可以根據經驗在一定時間內噴射一個區域進而達到預期的效果。

4.2.2 噴砂泵浦

它有一個上膠的鑄鐵葉輪和外殼，雙滑環密封，以及兩段變速馬達(10/3 kW)。泵從載體容器中吸引，壓力可皆由通過一個分離的閥門引導到每個噴槍，如圖 4-6 所示。泵浦也使用於淨空噴砂系統，然後將流動引導到噴砂艙下的過濾系統。

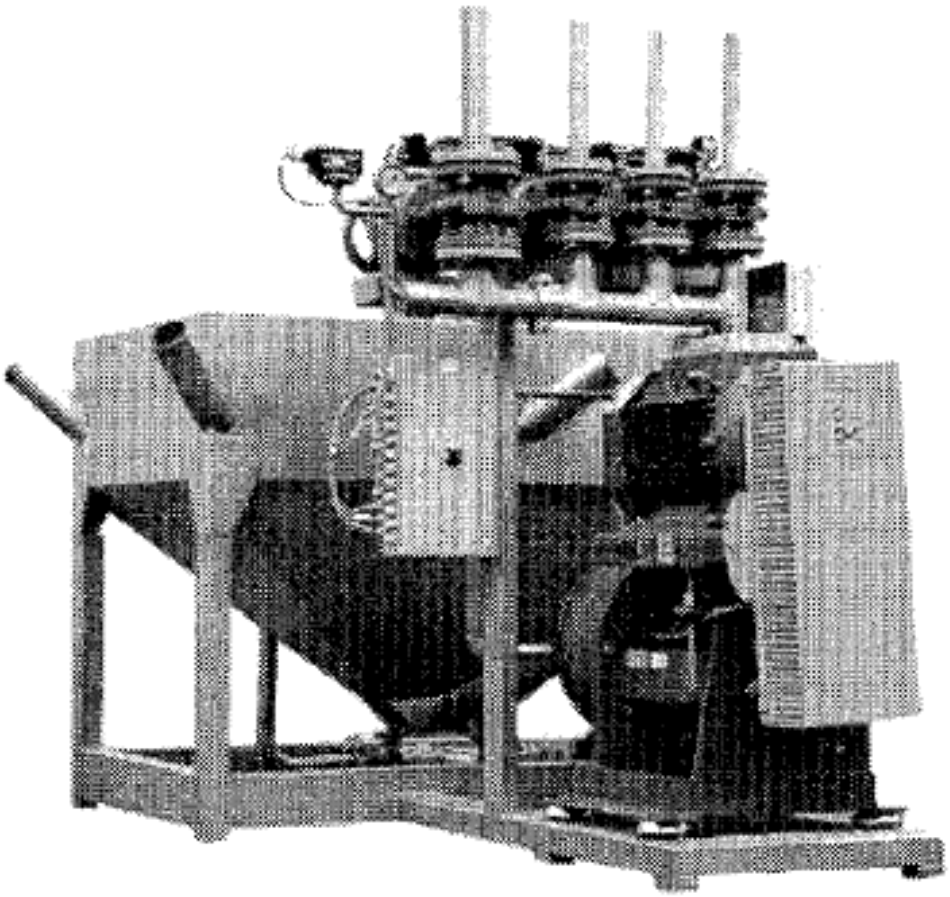

圖 4-6 噴砂泵浦系統[20]

4.2.3 載體和分離容器

在操作的期間，噴砂載體和水會通過載體容器進行循環，如圖所示，其運作方式也如同在操作期間的貯存槽。在開始的過程，會有一藉由載體混合水循環並與壓縮空氣結合的可能性。通過混合器內的螺旋槳進行額外的攪拌，該混合器螺旋槳會破壞噴砂載體最後的沉積物。過量的水和最終的粒子會溢出到分離容器中。該系統將溢流的噴砂載體連續送回到載體容器內，較輕的顆粒會溢流到淨空泵浦以傳輸到過濾系統。

4.2.4 濕式噴砂程序

當放射性物質或元件要在濕式噴砂系統中去除污染時，首先必須使用一正常的高壓水清洗。這將會移除所有沒附著在氧化層上的顆粒，且在一個封閉的噴砂艙來執行這項程序。由於噴砂載體的尺寸與被噴物質的幾何形狀彼此間是有相依關係的，他不是從一個外部操作位置上噴出，就是從操作員在噴砂艙內進行操作噴砂的工作。如果操作員在噴砂艙內，他們將穿著特殊的防護服，並會有新鮮空氣以供應，如圖 4-7 所示。在噴砂艙內是維持負壓的狀態，主要目的是為了確保不會有任何物體會從噴砂艙內流出到外界，而負壓條件則是會由噴砂艙對外部風扇系統做一連結，以維持其負壓條件。載體容器會裝有玻璃載體以及經過噴砂循環後的流體與載體，而在噴砂過程中玻璃載體則會開始慢慢產生變質的現象。這將導致噴砂效果降低。

一旦操作員根據他的經驗將所需完成的表面做完噴砂處理，其將會用水洗滌以除去任何載體。如果操作員是在噴砂艙內執行工作，他們將使用在噴砂艙內的淋浴器來將自己身上所黏在防護服上的任何玻璃珠載體。噴砂艙內有一個自動沖洗系統，可以沖洗噴砂艙內部，以便可將將所有粘在天花板，牆壁或地板上的玻璃珠流沖返回載體容器內。

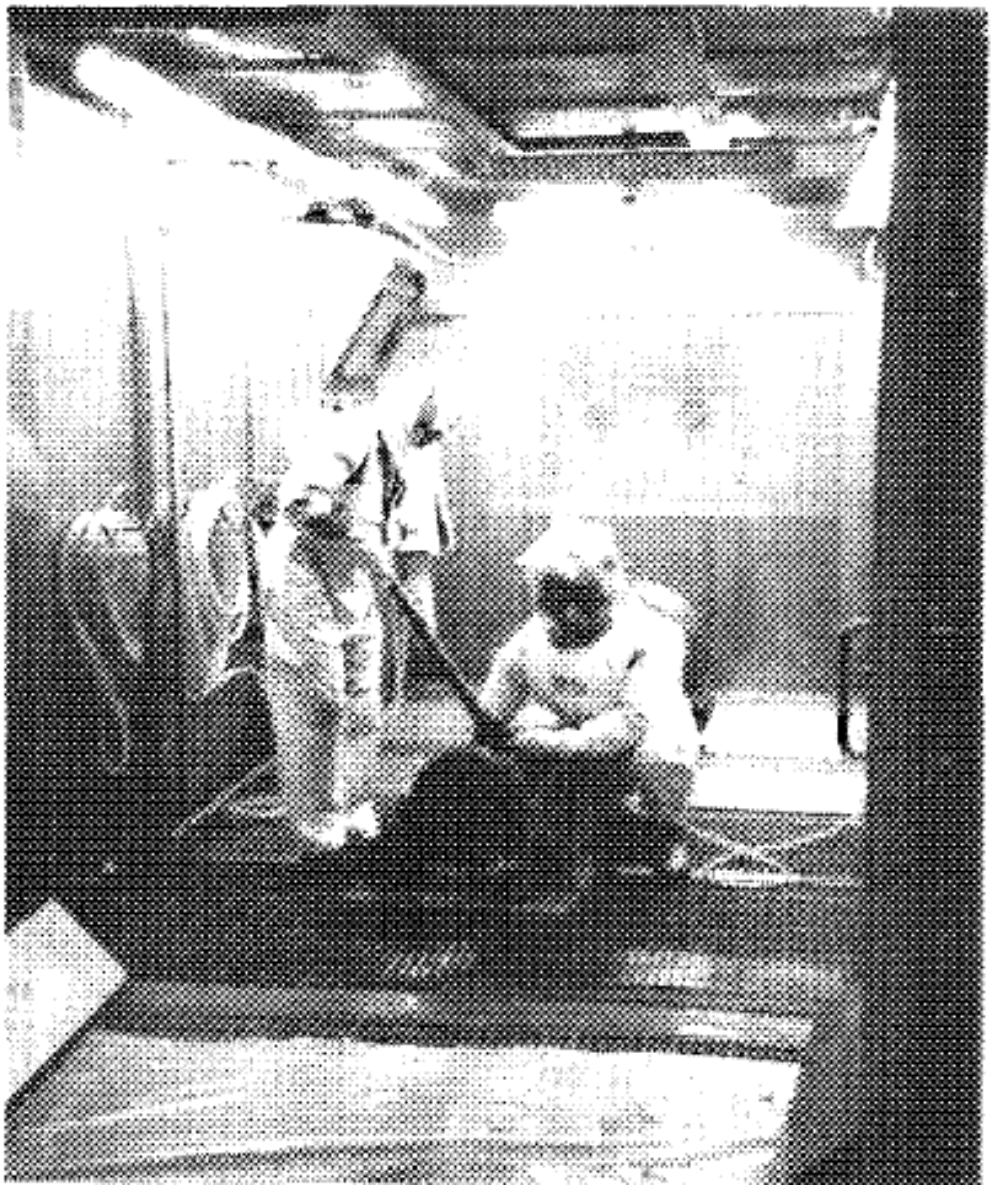

圖 4-7 操作員著特殊的防護服在噴砂艙內作業[20]

4.2.5 除污水處理

在噴砂系統中所使用的水是可以重新循環利用，儘管有過剩多餘的水，也可以回流至使用當作新鮮的水來進行清洗。這些水將會在載體容器內溢流且被泵浦輸送到電廠的水處理系統。在 BKAB，此系統會有其代號為 344，對於活性液體廢棄物的系統，此廢水將會依如圖 4-8 所示的此路徑來流動。

從除污部門水會經由泵浦輸送到沉澱池，且在這池中固體將會形成污泥。這種污泥會被泵浦輸送到一個污泥容器裡，在這容器內他將會有第二次機會變

成沉澱物。污泥容器裡的固體會被泵浦輸送到混凝土廢棄物處理容器，這樣的廢棄物被歸類為中等活性廢棄物，且將儲存在 SFR。污泥罐中的固體將被泵入混凝土中廢物處理容器。這種廢物被歸類為中等活性廢物存儲在 SFR。在沉澱和污泥容器之後，水則會被泵浦輸送到過濾裝置，然後測試其放射性。一旦確定放射性已被清除，則將會被抽回進入大海。

在濕式噴砂中被泵浦輸送到系統 344 所使用的水不是會被淡水沖洗就是被載體容器給清空。當每次從載體容器內約 250 公升的水被泵浦輸送到系統 344 時，所使用的玻璃載體大約每年會更換 10 次。對於清洗所使用的水將會溢流到載體容器以及被泵浦輸送至系統 344。使用在濕式噴砂的水總量，估計每年約為 150m3。供輸至系統 344 與其他系統的沉澱池保有 120m3，且每年約會清空 80 次，總計每年會有 9,600m3，每年這樣的水量會產生污泥量約為 5.5m3。如果來自除污部門的水如同從其他系統為受汙染的水，從濕式噴砂系統所帶來的污泥體積量為 150/9600x5.5m3=0.086m3 或 86 公升。由於濕式噴砂處理後的水除污過程所致，這將是每年需要在 SFR 貯存的污泥體積。根據資料庫確認檢查後顯示，結果證明有在超過一年的期間內，在 BKAB 的濕式噴砂處理後有 791 個項目已經被除污了。如下表為 BKAB 所顯示的資料。

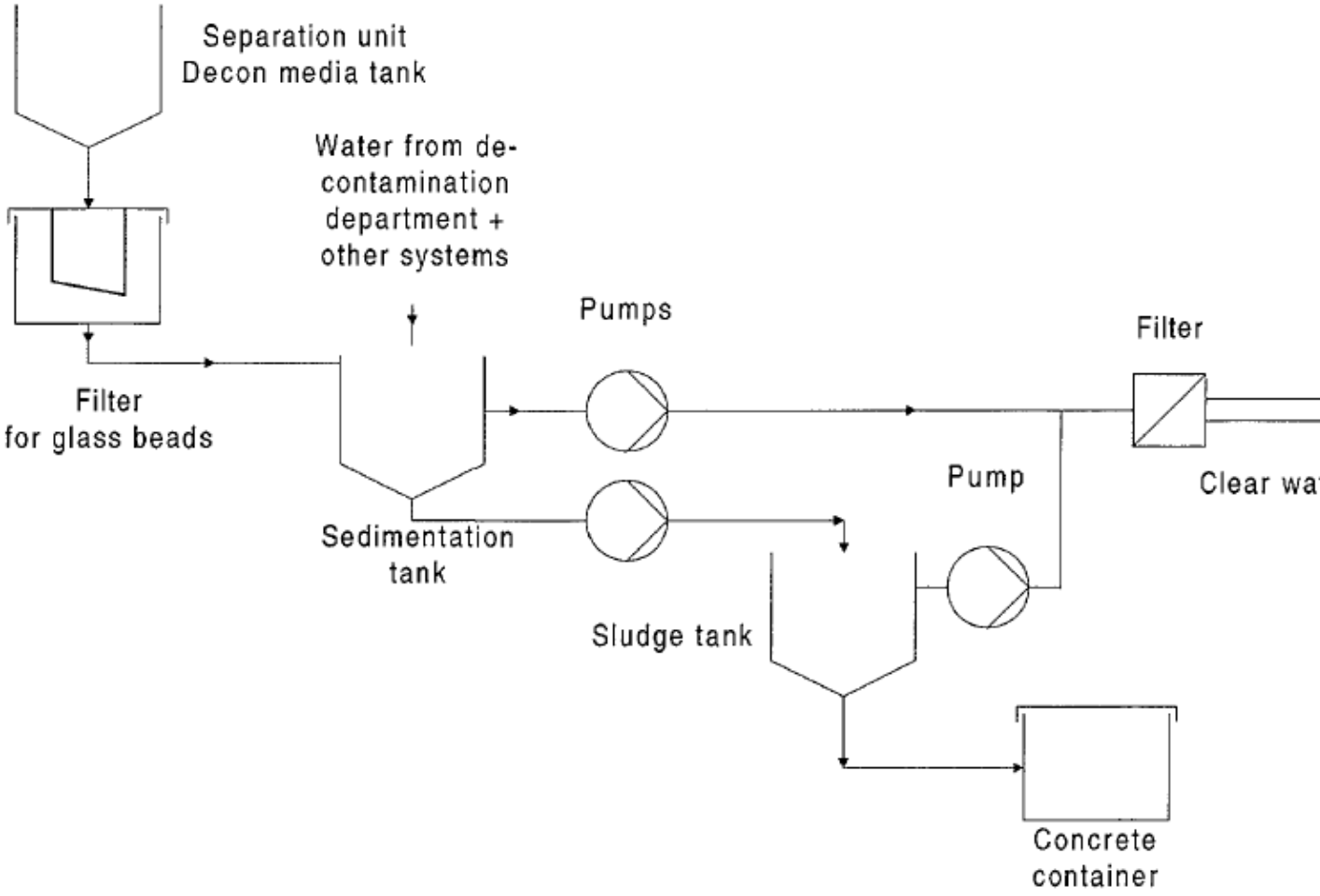

圖 4-8 除污水處理流程[20]

表 4-5 BKAB 的數據[20]

Part	No.	Date	Dosr before mSv/h	Dosr after mSv/h	Surface before Bq/kvcm	Surface after Bq/kvcm	Blasting	De-classification	Ellectropolishing
VALVE	1	9905	15	2.5	4250	4	X		
WEDGE	1	9905	60	9	1000	10	X		
VALVE	3	9905			10	2	X		
VALVE PARTS	2	9905	0.05	0.02	200	2	X		
WEDGE	1	9905	6.5	2	1000	4	X		
PUMP PARTS	2	9905			10	2	X		
PUMP PARTS	2	9905			10	2	X		
CRDM PARTS	500	9812	1.5		2000	2	X		X
WELDING EQUIP.		9811			12	0.3	X	X	
LATHES	3	9811	0.06		20	0.3	X	X	
INDUCER	1	9811			30	1	X	X	
SHAFT	1	9811	12	0.6			X		
IMPELLER	1	9811	40	2			X		
CUTTING DEVICE	250	9811	1		1000	0.3	X	X	
IMPELLER	1	9810	65	8	4000	4	X		
PUMP PARTS	4	9810			2000	20	X		
IMPELLER	1	9809	18	4	4000	3	X		
HEAT SHIELD	1	9809	14	4	3000	10	X		
HOUSING	1	9809	20	1	3000	20	X		
STEM	1	9809	40	1	3000	20	X		
WEDGE	1	9809	80	20	3000	20	X		
PULL ROD	2	9809	100	0.5	3000	20	X		
WEDGE	1	9809	50	10	3000	20	X		
STEM	1	9809	25	4	3000	20	X		
HOUSING	1	9809	22	1	3000	20	X		
PULL ROD	2	9809	100	0.5	3000	20	X		
VALVE PARTS	1	9809			10	2	X		
PUMP PARTS	2	9807	0.01	0.01	16	2	X		
SPRINGS	3	9805			5	2	X	X	

4.2.6 使用玻璃噴砂處理之經濟性

放射性廢棄物可以分為許多不同的類別。當它與濕式噴砂有關的廢棄物時，廢棄物的活性將會是中、低標準的放射性廢棄物。當瑞典核能計劃決定在 Forsmark 的廢棄物儲存設施建造時，其是為了能容納中、低標準之放射性廢棄物。建立這項設施以及利用整個核能計畫的一部分來支付，並且每個電廠均擁有一部分的貯存空間。此貯存空間主要是由一部分低放射性標準以及中放射性標準為一個單位所組成。低放射性標準廢棄物貯存在大型岩石拱頂的標準容器中，因為此廢棄物放射極低劑量標準的輻射，他們可以使用正常正常的堆高機來處理。而中等放射性標準之廢棄物的貯存與處置在一個特殊的混凝土容器中，每個容器的體積為 1.44m^3，其中 0.3m^3 可用於廢棄物的處置。混凝土容器可以利用遠端遙控來處理但不需要冷卻。貯存的成本將可根據新設施的成本來評估貯存廢棄物，已存在的設施即將填滿，且今日可除役的廢棄物將可儲存於未來的成本上。

這份報告也提及必須要記得濕式噴砂的主要目的是能以最大限度減少電廠工作人員的劑量。然而，濕式噴砂設備可以在電廠隨時有系統地使用，也允許 ignots 可以在 20 年內被回收。濕式噴砂也可以在沒有任何劇烈變化的情況下轉換使用更具研磨性的噴砂介質，如鋁矽石。此將允許部分元件已被除污並從放射性物料名單中剃除，進而使用一般通道來回收。藉由增加介質的有效性，所需噴砂的時間將會縮減。縮短每項物品對操作員的暴露時間。

附錄一有詳細的報告可供參考。

4.3 噴砂除汙對除汙作業方法之效益

採用噴砂除污技術一般均須建立大型帳棚防止污染擴散，以利後續進行除污完成後之輻射偵檢。噴砂除污技術屬於機械方式之除污，在除役的規劃與實際執行時，必須列入為二次廢棄物管理，此重點在於噴砂除污作業可能衍生的二次廢棄物及必須進行減廢措施。乾式噴砂研磨時，須在工作區域加裝粉塵控制系統及真空過濾系統，以降低粉塵污染狀況。而使用濕式噴砂研磨時，會產生大量的廢水、磨料及研磨後的碎屑粉塵，過程中必須要採取適當的減廢措施。因此，在設備考量與成本上將會有所增加，且二次廢棄物的處理與技術的樣式與種類也必須考量二次廢棄物處理之難易度、成本與適用性。

從國際上的噴砂除污技術經驗來看，雖然成效上來看是可以接受，但是所增設的設備與成本、二次廢棄物處理等又是一項需額外考量的地方。因此，以目前可行的除污方式與國內進行除污技術的選擇上，噴砂除污技術是否有必要增列至核一廠除役時進行，其實可以列為備用方案，以其他除污方式，如化學除汙，做為主要除污方式。

第五章 結論

本次期中報告共分五個章節，除了首章和次章的前言和計畫目標之外，於第 3 章對本次計畫期程進行了研究工作進度概述，首先概要說明除役核電廠的除污技術概述，內容包含核電廠除役作業階段、機械除污技術說明，以及審查除役計畫時應注意的項目。第四章則說明噴砂除污技術實際應用的情況。

在本計畫中依除役除污作業的實際案例進行資料的蒐集與研讀，著重於噴砂除污技術的應用與做法，了解國外核電廠除役時所採用的除污方法。由於國內並無除役的實質經驗，針對國內首度將進行除役之核電廠必須以最嚴謹的安全考量為基礎下，針對噴砂除污技術應用於國內核一廠的重要性並亦須具備完善之應變計畫與方法，以及防止意外發生與降低意外災害。瑞典在核電廠除污方面，認為濕式噴砂除污方式相較於其他的方法有較好的效果，且廢棄物也較容易處理與貯存。不過噴砂除污技術需要額外增加設備與二次廢棄物處理，整體成本也會有所增加。因此，核一廠在進行除污時，可將噴砂除污技術列為備用方案。

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附錄一 瑞典電廠於濕式噴砂應用文獻

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Study of wet blasting of components in nuclear power stations

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Study of wet blasting of components in nuclear power stations

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This report looks at the method of wet blasting radioactive components in nuclear power stations. Fagerström Industrikonsult AB and their division DECO Systems supply the wet blasting cabin in focus. These cabins are today in operation at Barsebäck Kraft AB and Oskarshamns Kraft Grupp.
The wet blaster uses pearl shaped glass beads with the dimensions of 150-250 µm mixed with water as blasting media.. The improved design, providing outer operator's positions with proper radiation protection and more efficient blasting equipment has resulted in a lesser dose taken by the operators.
The main reason to decontaminate components in nuclear power plants are to enable service on these components. On components like valves, pump shafts, pipes etc. oxides forms and bind radiation. These components are normally situated at some distance from the reactor core and will mainly suffer from radiation from so called activation products. All nuclear power plants in Sweden have special decontamination departments to decontaminate radioactive components.
When a component is to be decontaminated it can be decontaminated to a radioactive level where it will be declassified. This report has found levels ranging from 150-1000 Bq/kg allowing declassification of radioactive materials. This difference is found between different countries and different organisations.
The report also looks at the levels of waste generated using wet blasting. This is done by tracking the contamination to determine where it collects. It is either collected in the water treatment plant or collected in the blasting media. At Barsebäck the waste levels, from de-contaminating nearly 800 components in one year, results in a waste volume of about 0,250 m3. This waste consists of low and medium level waste and will cost about 3 600 EURO to store.
The conclusions of the report are that wet blasting is an indispensable way to treat contaminated components in modern nuclear power plants. The wet blasting equipment can be improved by using a robot enabling the operators to remotely treat components from the outer operator's positions. There they will benefit from better radiation protection thus further reduce their taken dose. The wet blasting equipment could also be used to better control the levels of radioactivity on components being sent for melting

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Johan Hall

Summary

This report looks at the method of wet blasting radioactive components in nuclear power stations. Fagerström Industrikonsult AB and their division DECO Systems supply the wet blasting cabin in focus. These cabins are today in operation at Barsebäck Kraft AB and Oskarshamns Kraft Grupp.

The wet blaster uses pearl shaped glass beads with the dimensions of 150-250 µm mixed with water as blasting media. This size of glass beads allows blasting on metal surfaces without, from a functional consideration, damaging the surface or structure of the material. Due to improved design and being larger the cabins facilitate decontamination of larger components. The improved design, providing outer operator's positions with proper radiation protection and more efficient blasting equipment has resulted in a lesser dose taken by the operators.

The main reason to de-contaminate components in nuclear power plants are to enable service on these components. On components like valves, pump shafts, pipes etc. oxides forms and bind radiation. These components are normally situated at some distance from the reactor core and will mainly suffer from radiation from so called activation products. Activation products are small fragments of metal, which is loose in the reactor system, and when they pass the reactor core they are exposed to neutron radiation and become radioactive. The components, which in some cases are highly radioactive, are handled by specialists at the in-house decontamination department where the components are washed and treated to remove the oxide bound radiation. All nuclear power plants in Sweden have special decontamination departments to de-contaminate radioactive components.

When a component is to be de-contaminated it can be de-contaminated to a radioactive level where it will be declassified. If the component can be declassified it can be used freely outside the power plant. In Sweden and internationally the level at which declassification is granted varies. This report has found levels ranging from 150-1000 Bq/kg allowing declassification of radioactive materials. This difference is found between different countries and different organisations.

Wet blasting is one way to remove the oxide bound radiation that is found on components in nuclear power stations. When compared with the other main options available, electro polishing and chemical bating, wet blasting provides comparable or better decontamination factors leaving a derived waste easier to handle and store. The operator also takes less dose per item using wet blasting than electro polishing as a result of longer exposure times close to the components when fitting the electrodes.

The report also looks at the levels of waste generated using wet blasting. This is done by tracking the contamination to determine where it collects. It is either collected in the water treatment plant or collected in the blasting media. At Barsebäck the waste levels, from de-contaminating nearly 800 components in one year, results in a waste volume of about 0,250 m^3. This waste consists of low and medium level waste and will cost about 3 600 EURO to store. This cost includes the storage volume of the concrete containers in which the waste is stored and handled. The Waste is stored at Forsmarks storage facility for the Swedish low a medium level radioactive waste.

The conclusions of the report are that wet blasting is an indispensable way to treat contaminated components in modern nuclear power plants. The wet blasting equipment can be improved by using a robot enabling the operators to remotely treat components from the outer operator's positions. There they will benefit from better radiation protection thus further reduce their taken dose. The wet blasting equipment could also be used to better control the levels of radioactivity on components being sent for melting.

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Appendix A: Governing Organisations

Appendix B: Specific Radio Nuclide Guidelines, EU

Appendix C: Data from Barsebäck Kraft AB

Study of wet blasting of components in nuclear power stations.

1 Introduction

1.1 Background

In the nuclear industry many parts are being contaminated as a result of their exposure to radioactivity. On some of these parts periodic service is performed and decontamination is therefore required to minimise the exposure to radiation to the service personnel. The radioactivity on these parts is in most cases contained in the oxide layers formed on the surface of the part. By removing the oxide layer using wet blasting the radioactivity of the part can be reduced to a level where the radiation dose received by the personnel is reduced to an acceptable level.

When parts are to be decommissioned from a radioactively controlled area they have to be handled and stored in agreement with current regulations on handling of radioactive waste. This handling and storage are a costly procedure and by using wet blasting some of the parts could be cleaned to de-classify them from the requirement of radioactive storage.

Fagerström Industrikonsult AB is a company with many years experience from decontamination of radioactive components in nuclear power stations. Among their range of products is the wet blasting cabin, from the Deco Systems division, that is in focus in this report.

1.2 Purpose

There has been a consensus among the policy makers in the Scandinavian nuclear industry that wet basting is a good way of handling contaminated parts but so far no one has looked at the various levels of waste products generated in this process.

The purpose with this report is to look at the process of wet blasting and the waste levels generated and to compare the volume and radioactivity of the waste versus the treated parts. And to look at guidelines issued by international organisations for declassification of radioactive materials.

1.3 Method

The information for this report has been gathered through interviews with personnel at Swedish nuclear power station's decontamination departments, governing nuclear institutions, associated industries and literature studies.

1.4 The presentation of this report

For the reader to easily follow this report it starts with a presentation of the power stations from which the bulk of information has been collected. It thereafter presents the wet blasting equipment followed by regulations and definitions of radioactive waste. The various forms of waste will then be examined to see where they collect as a result of wet blasting. How well the wet blaster performs is then studied with the help of data from the power stations. After which the finding of the report is concluded.

1.5 Two nuclear power plants and a storage facility

1.5.1 Barsebäck Kraft AB, BKAB

Barsebäck Kraft AB is the owner and operator of the Barsebäck Nuclear Power Plant and a wholly owned subsidiary company of Sydkraft AB. Sydkraft is a private energy company,

operating in a competitive market. The three largest shareholders are foreign they are, Statkraft in Norway, PreussenElektra and Hamburgische Electricitäts-Werke in Germany. The plant has two BWR-units, with a capacity of 615 MW e each. They were commissioned in 1975 and 1977. Barsebäck one was shut down on the 30'th of November 1999 and will be dismantled in the years to come. One decontamination department service both reactors and they have two blasting cabins. One smaller, which has been in service since 1986 and one larger unit supplied by Fagerström Deco Systems in year 1998.

1.5.2 Oskarshamns Kraft Grupp, OKG

Three power corporations owns Oskarshamns Kraft Grupp, they are Sydkraft AB, Birka Energi AB and Stora Enso.

The plant has tree BWR-units, O1, O2 and O3 with a capacity of 445 MW e, 605 MW e and 1160 MW e. The first reactor O1 was also the first commercial reactor in Sweden and commissioned in 1972, O2 in 1974 and O3 which, was the last reactor to be commissioned in Sweden, in 1985. One decontamination department service all three reactors and they have two blasting cabins. One smaller and one larger unit supplied by Fagerström Deco Systems in year 1997.

1.5.3 The Final Repository for Radioactive Operational Waste - SFR

The Final Repository for Radioactive Operational Waste is located at Forsmark's nuclear power plant, 50 meters under the Baltic Sea level. All the low-, and intermediate-level short-lived waste from the nuclear power plants is stored at SFR. Low-level waste is placed in four rock vaults, each a length of 160 metres. The intermediate-level waste is stored in packages and placed in a silo, surrounded by protective barriers.

2 The wet blasting cabinet and how it works

2.1 The wet blasting cabinet and auxiliary equipment

In Sweden there are four nuclear power plants, Barsebäck, Oskarshamn, Ringhals and Forsmark. They all have a decontamination department using, among other methods, wet blasting as a way of decontamination.

Barsebäck and the decontamination department at Oskarshamn both have similar blasting cabins supplied by Fagerström Deco Systems. See figure 1.

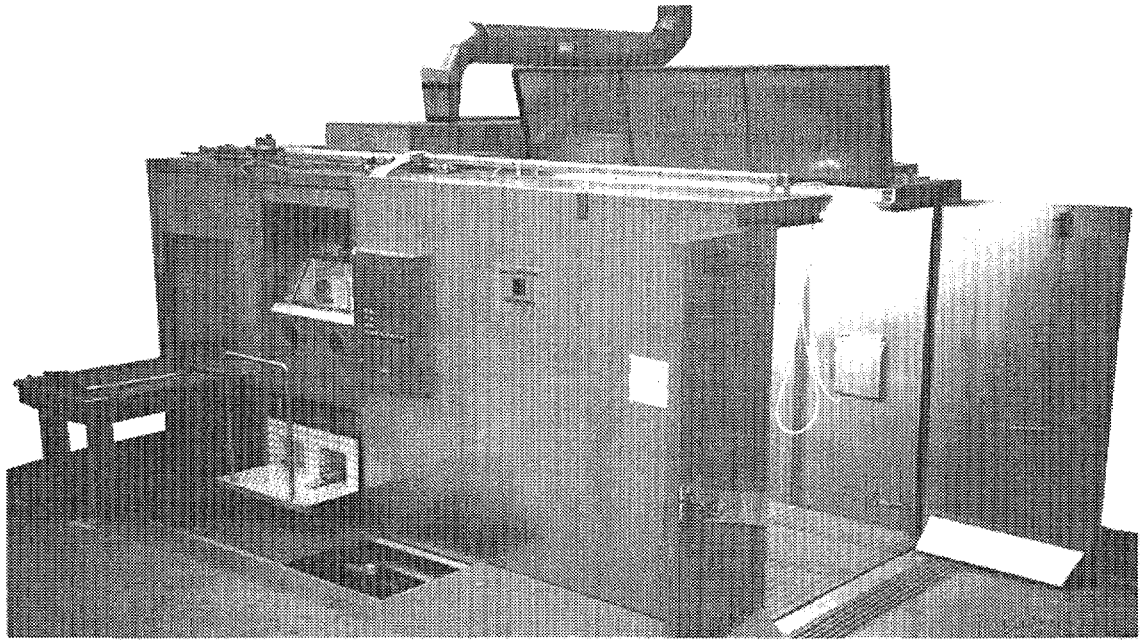

Figure 1.

The wet blasting of components is done in a tight cabin, with inner dimensions of 5500 mm x 3400 mm x 2500 mm. Components with a maximum weight of 2000 kg may rest on the support beams. The grating is dimensioned for a load of 500 kg/m2. A double door is situated on one of the short sides of the cabin and together with a roof hatch allows transportation into the cabin by crane.

The cabin itself is one unit. Media pump, media tank, separation tank and emptying pump are assembled as another. The filter handling system is also an individual unit. To meet the requirements of the nuclear industry the wet blasting decontamination system is manufactured out of cold rolled stainless steel, type 304 surface class 2B high finish. Media tank and the pipe systems are manufactured in acid proof stainless steel, type 316. Certain components as the media pump and media valves have rubberised inner surfaces. All external joints have flush proof lists and the roof is tightly welded to enable external washing. To minimise noise the cabin is sound insulated.

Besides one of the radiation protected operating positions a guillotine hatch intake is situated. This hatch is operated by a switch and equipped with an automatic jam safety device. If any obstacle is in its way during closing, the hatch moves up. A rotating table can be pulled out of the cabin to an external position to facilitate loading and unloading. This external position is equipped with a water nozzle for rinsing of components. From his position outside the cabin the operator can remotely manoeuvre the rotation of the table in the either direction. The turntable's start is smooth and accelerates to a pre-set point of velocity. The cabin is, on the short side, internally divided with a four-mm transparent foldable plastic curtain allowing the operator to minimise the spread of particles in the cabin.

The wet blasting system is designed for simultaneous use of the three blasting nozzles. The lower part of the cabinet, under the grating, is cone shaped in six separate sections each ending over a hose transporting the media back to the storage media tank. All cones are equipped with a nozzle rinsing system for cleaning. The Wet Blasting Decontamination System has two outer operators positions, which are radiation protected with 50 mm lead (1280 kg) and 76-82 mm lead glass windows (equivalent to 24,9 mm lead). Both positions have oval glove penetrations.

2.2 Blasting material

The blasting material used is glass and it is used in the shape of pearls. It has been proven that glass pearls has a polishing effect on harder surfaces and hardly any degrading of the surface, as will abrasive materials such as sand.

Glass pearls used for blasting can be bought in different gradings. But the grading found most suitable for blasting off oxide layers on hard metal is the 150-250 µm. The glass pearls will degrade over time but this will not result in a degrading of the surface. If larger gradings are used it may result in peening which is a hardening of the surface of the metal. This may not be desirable and therefore the grading size of 150-250 µm is used, as it does not result in any measurable hardening of the surface.

The experience from the use of wet blasting equipment is that it is a method that can be used to remove the oxide layers without impairing the components when using the right size of glass pearls. The operator will, for a certain time, based on experience, blast an area to achieve the desired result.

2.3 Blasting pump

A goods pump drives the blasting media. It has a rubberised cast iron impeller and housing, double slide ring seal, two-speed motor (10/3 kW). The pump sucks from the media tank and the pressure is guided to each blasting gun through a separate valve, see figure 2. The pump is also used for emptying the blasting system when it is time to change the blasting material. The flow is then guided to the emptying cyclone, which is connected to the filter holder under the cabinet.

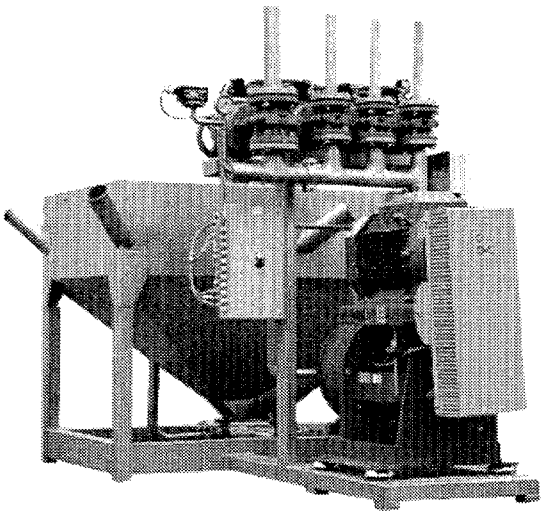

Figure 2.

2.4 Media and separation tank

During operation, blasting media and water is circulated through the media tank, see figure 2, which also works as storage between the operation periods. At the starting sequence, there is a possibility to mix the media through circulation of water combined with compressed air. Additional stirring is performed through a mixer propeller, which breaks eventual sediments of blasting media. Excessive water and eventual particles overflows to the separation tank. The system has continuous feedback of overflow blasting media to the media tank. Light particles overflow to the emptying pump for transportation to the filter unit.

2.5 Emptying blasting media and water

The glass beads normally resists a large number of blasting cycles before they are used up. It is mainly the collection of radioactive material among the beads that decides the time for

emptying. During emptying, beads and water is pumped via the cyclone to the filter container. The cyclone causes the heavy particles, glass beads and blasted off oxide, to land in the filter sack and prevents pressure building up. When the system is totally evacuated the pump automatically stops when loss in suction ability is detected.

2.6 Emptying, waste handling of filter sack

At BKAB a good operator's environment is ensured through a remote controlled filter-handling system, which minimises radiation exposure. The emptying system is designed for standard 200 litre shielded barrels but can be customised to fit other barrels.

3 How to measure radiation and levels of radioactivity for declassification in Sweden and Internationally

3.1 Measuring and monitoring ionising radiation

The human senses cannot detect radiation or discern whether a material is radioactive. However, a variety of instruments can detect and measure radiation reliably and accurately. A radioactive source is characterised by its activity that is stated in the unit Bequerel. One Bequerel means that one radioactive atom nucleus deteriorates per second. Ionising radiation is measured in the international unite, the Gray (Gy) and the Sievert (Sv).

The amount of radiation, or 'dose', received by a person is measured in terms of the energy absorbed in the body tissue, and is expressed in Grays.

Equal exposures to different types of radiation do not however necessarily produce equal biological effects. One Gray of alpha radiation, for example, will have a greater effect than one Gray of beta radiation. When we talk about radiation effects, we therefore express the radiation in units called Sieverts. In table 1 an overview is showed.

Table 1. [1]

Magnitude	Unit		Definition
	Name	Symbol	
Activity	Bequerel	Bq	1 Bq = 1 deterioration/s
Absorbed dose	Gray	Gy	1 Gy = 1 joule/kg
Dose equivalent	Sievert	Sv	1 Sv = 1 joule/kg

The various types of radiation we may be exposed to have as mentioned different biological effects. The quality variation can be found in the table 2.

Table 2. [1]

Radiation type	Quality factor
X-ray, γ- and β- radiation	1
Neutrons, thermal	2-3
Neutrons, other	10
α- Radiation	20

As seen in table 2 alpha radiation has 20 times greater biological effect than gamma radiation. Due to the difference in penetrating power of the various types of radiation different types or radiation protection is required to stop the radiation. The different penetrating capability of various radiation types is shown in figure 3.

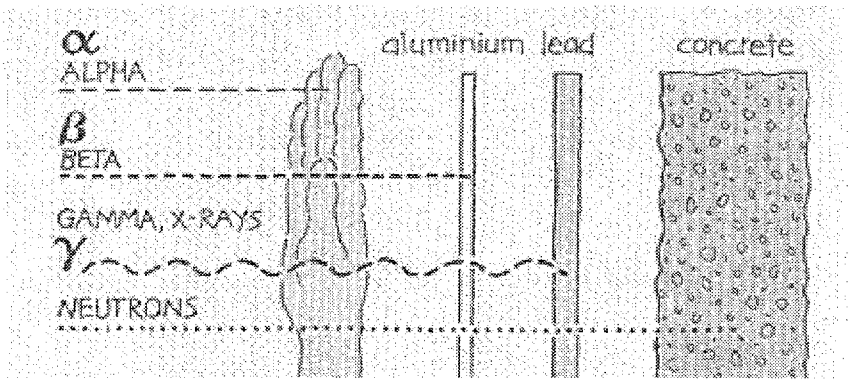

Figure 3.[2]

3.2 Authorization and clearance for disposal, recycling or reuse

3.2.1 Regulations in Sweden

There are many different levels and guidelines proposed by various organizations regarding the levels of radiation accepted for clearance for disposal of radioactive waste.

In SSI FS 1996:2 the Swedish Radiation Protection Institute have issued regulations regarding clearance of goods from the Swedish nuclear facilities. The following paragraphs presents the regulations relating to solid materials:

§8 The surface contamination on goods to be taken out of zone divided area may not exceed[1]

1. 40 kBq/m^2 total of Beta- and Gamma radiating nuclides.
2. 4 kBq/m^2 of Alpha radiating nuclides.

The surface contamination is to be calculated as an average of 0,03 m^3.

Free usage

§9 When exporting goods from a zone divided area for the purpose of free usage the goods may contain at most 500 Bq/kg of which no more than 100 Bq/kg may derive from Alpha radiating nuclides. These values are the value exceeding the natural activity of the goods when present outside a nuclear facility. The limiting value need not be applied to each individual object such as occasional tools, which are brought out. However they shall under these circumstances comply with the surface contamination as specified in §8.

Deposit of waste on private or council waste disposal sites

§10 When exporting goods from a zone divided area for deposit on private or council waste disposal sites, the goods may contain at most 5kBq/kg of beta- and gamma radiating nuclides and 0,5 kBq/kg of alpha radiating nuclides. These values are the value exceeding the natural activity of the goods when present outside a nuclear facility.

[1] For goods in particle form, e.g. blasting material, it is sufficient to fulfil the threshold value for specific activity in accordance with §9 and/or §10.

3.2.2 International regulations

If we look to the international community there are a number of organisations that have issued guidelines for levels of radioactivity in materials and tools to be declassified. In Appendix A regulations from the following organisations can be found: IAEA (International Atomic Energy Agency), OECD/NED (Organisation for Economic Co-operation and Development/Nuclear Energy Agency), EU-Commission (European Union), and Nordic Nuclear Safety Research Project. These guidelines will give an indication on the levels for which clearance are given in the European countries.

In Appendix B: excerpts from the European Community Radiation Protection Legislation Document can be found. In this document a comprehensive list for both quantity and concentration of radioactivity, for each individual radionuclide, can be found. This list can be used for objects where one have measured activity for each specific nuclide and wish to know their specific acceptable limits.

If we take one common nuclide in oxide waste, Cobalt 60, and compare the organisations above and their proposed levels of activity for declassification we get table 3.

Table 3.

Nuclide	**IAEA**	**OECD/NEA**	**SSI/Sverige**	**Finland**
Cobalt 60, β-radiator	300 Bq/kg	150 Bq/kg	500 Bq/kg	1000 Bq/kg

As seen in table 3 there are some considerable variation in the levels regulating activity in goods for declassification from nuclear power plants. Each country has specific regulations governing the legal level of activity.

3.3 Levels and volumes for storage

The activity levels are as previously described regulated on activity and specific radionuclide levels. At OKG the metal waste such as scrapped components and tools can, depending on their activity level, be sent to various places for disposal. The components/tools are first cleaned after which they are checked for surface contamination. The components/tools level of contamination will classify them. They are then either recycled or deposited. The components/tools will be measured for radioactivity and then treated according to the activity level, se table 3.

Table 3. [3]

Type of handling:	**Activity level:**
Scrap metal for re-use	< 0,5 kBq/kg
Council waste site, 1 GBq/year	< 5 kBq/kg
OKG's ground deposit site, 100GBq	<300 kBq/kg
Deposit at SFR	>300 kBq/kg

3.4 Different radio nuclides present

In a nuclear power plant many different types of radionuclides are present. When looking at decontamination of contaminated parts in a power plant the ones to pay attention to are found in table 4.

Table 4. [4]

Radio nuclide	
Fe-55	Sb-125
Co-58	Cs-134
Co-60	Cs-137

Ni-63	Eu-152
Zn-65	Pl-238
St-90	Pl-239
Zr-95	Pl-240
Nb-93	Pl-241
Ri-106	Am-241
Ag-110m	Zm-244
Sn-113	

The radio nuclides listed in table 4 are all present in the reactor systems of a nuclear power plant and the ones regarded as the ones to keep watch on when it comes to storage of radioactive parts over a time period of about 20 years.

The bulk of active products are the ones that are metals from Fe 55 to Zn 65. They appear as a result of small fragments of lose metal in the system that passes through the reactor core where they receive neutron radiation and becomes activated, so called activation products. In a reactor system the release of metal from valves etc. is between 50-100 grams/year. The activated products are β and γ-radiators who carries a quality factor of 1.

Radioactivity in the reactor coolant system (RCS) can be attributed preliminary to the presence of the isotopes Cobalt-58 and Cobalt-60, which are derived from the elements Nickel-58 and Cobalt-59 via neutron interaction. Nickel-58 is a constituent of austenitic steels and Cobalt-59 is a major alloying element used in-wear resistant hard facings (Stellite®)

Alpha radiation that carries a higher quality factor of 20 is normally only found in nuclides with atom number higher 210. As seen in table 4 there are a number of nuclides with atom number higher than 210 but in order for these products to be present in, any substantial amount, there must have been a release of materials from the fuel.

In some of the Swedish reactors there has been damage to the fuel cells, which has resulted in leakage of fuel into the reactor water with the result of contamination by Alpha emitters to the system. In these cases one has to take into account for Alpha-radiating nuclides, as they will now be part of the system and often carries long half times.

Knowing which types of nuclides are present in the reactor system is of interest when it comes to treatment of components prior to storage. One way of handling components is to melt them. This handling will reduce the level of radioactivity by vaporising as well as redistribution in the metal. If the radioactive level can be reduced to about 15-16 Bq/gram [5] prior to melting and the bulk of this radiation is the result of the nuclide Cobalt 60. The metal will be recyclable in 20 years and thus not needing storage at SFR that would otherwise be the case. This reduction rate could possibly be achieved using wet blasting.

3.5 Definition of radioactive waste

Radioactive waste is divided in separate categories depending on its activity. In Sweden the classification is Low, Medium and High activity waste. Other countries have similar definitions but they may vary to a certain degree.

How the waste is handled depends on in which form it is present and how great the activity content is.

3.5.1 Low level waste

The majority of the radioactive waste emits a very low level of radiation. It contains radionuclides with relatively short half-lives. This type of waste can consist of rags, old

overalls, used paper towels, plastic, tools, etc. and are collectively labelled low-level waste. Low-level waste can be transported and handled in simple packaging without special precautions.

All waste from zone divided areas is measured and sorted. Metal waste is often so low in activity that once it has been decontaminated it can be recycled in the metal industry.

The low-level waste is measured for activity, depending on the level of activity it is either shipped to SFR for storage or it may be stored at onsite storage sites.

If the waste is to be stored onsite[2] it is placed on a concrete slab that has an angle to facilitate collection of drainage water for control. The waste is covered by a plastic sheet and then covered by soil. After approximately 25 years the activity has decreased to a safe level.

3.5.2 Medium level waste

The majority of the medium level waste generated consists of filter material from the ion exchangers. These are used to clean the water in the reactor system, condensation system, fuel basins and the wastewater from the decontamination department. Other types or waste classified as medium level waste can be used piping, valves, machine parts and used protective clothing.

When working with medium level wastes some degree of radiation shielding is required. However no special cooling of the waste is required.

The filter material used for cleaning the contaminated water is cast in special container casks each with a capacity of 300 litres. Filter material with lower radiation levels is put in larger concrete tanks, each with a capacity of 5,5 m3, and drained of water.

Medium level waste such as rubbish and metal is cast in casks. All packages are measured on a nuclide specific basis and registered.

The purpose of the storage is to prevent leakage of radioactive nuclides for the first 500 years. After this time period the radiation levels will have decreased to a level where their contribution to the surroundings will be negligible.

3.5.3 High-level waste

High-level waste consists to the majority of used fuel. It is highly radioactive and develops large quantities of heat. It will therefore require cooling and radiation protection for ca 40 years before it is stored. The current proposal is to store the high level waste deep under ground.

4 Waste volume derived from wet blasting

4.1 Decontamination of radioactive parts and equipment

Many parts in a nuclear power plant need regular service or replacement. In order to render possible service the parts has to be de-assembled and taken to workshops. To minimise the radioactive dosage taken by the service personnel many of the parts exposed to radiation require some kind of cleaning. In Sweden and many other countries the ALARA principle (ALARA = As Low As Reasonably Achievable) is adapted. The purpose of ALARA is to minimise the dosage to a group of individuals. By de-contaminate a part before work is carried out on the part the collective dosage taken by the personnel involved is lowered.

[2] At OKG and Ringhals they have onsite storage.

Investments are continuously made in order to lower the dosage to the personnel involved in handling radioactive parts. If an investment in new equipment etc. can lower the dosage it will be done if the investment criteria can be met. It needs to be stressed that this is a criteria that exists as a guideline for investments. The current criteria for investment are 465 EURO per saved man mSv and year. I.e. if an investment in new decontamination equipment can save 10 people 5mSv/year each the total investment can be 465 x 10 x 5 = 23 250 EURO.

When a contaminated part is to be serviced or perhaps moved out of a controlled zone it may require decontamination if the radioactive levels are found to be too high. Depending on the part, which material it is made of, where in the plant it is situated, how severe the contamination is various methods of decontamination are used. On the flow chart below, see figure 4, we can see the options at hand in a modern decontamination department at a nuclear power station.

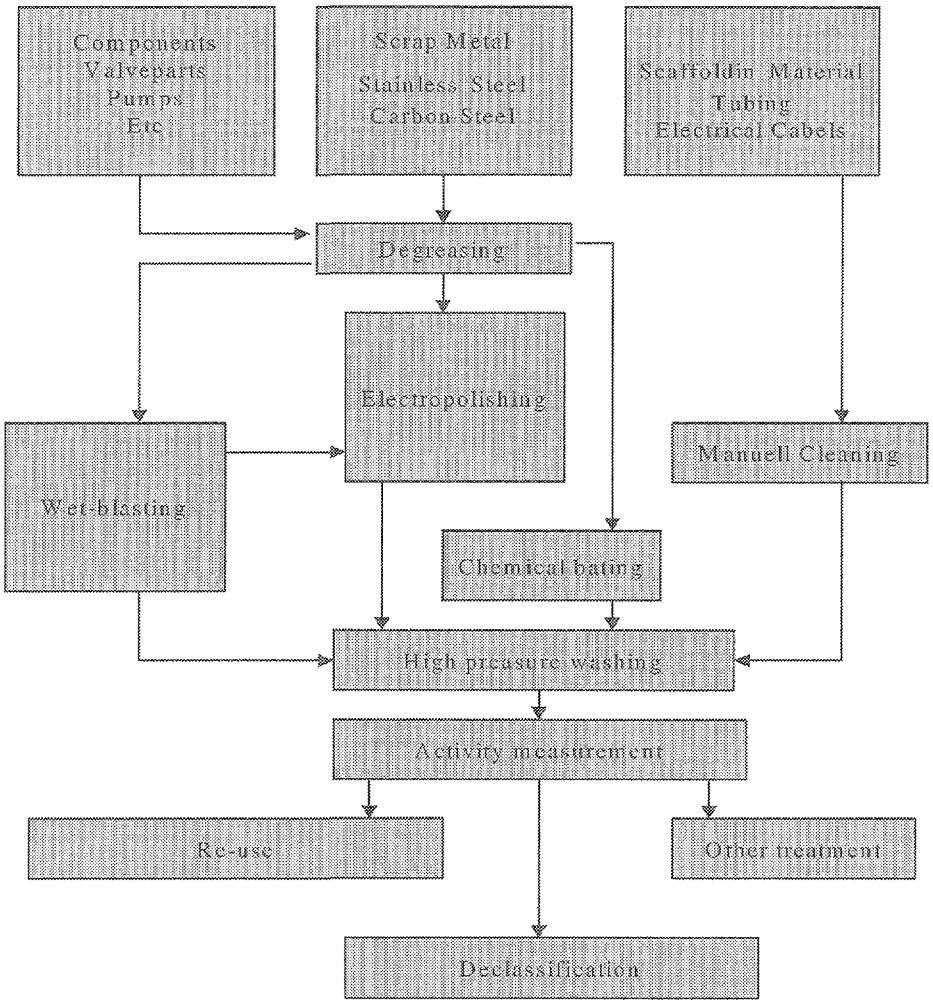

Figure 4.

The arrows also indicate in which order the various treatments will take place. The options available are:

- Degreasing, when parts have been subjected to their surrounding environment they require degreasing to remove any oil films present. This is performed in order to make the decontamination more efficient but also to spare the upcoming treatments from oil contamination.
- Electropolishing, decontamination of metal components is performed using an electrochemical process that removes the active elements bound to the surface layer. The object to be cleaned is put in a neutral electrolyte, alkali metal sulphate, and is connected to a rectifier. Metal will be removed from the object as metal ions. The object normally forms the anode (+) but can automatically be switched to form a cathode (-) for short intervals in order to increase efficiency. The metal ions will precipitate as a metal hydroxide sludge that is taken to a sedimentation tank where it can be separated from the electrolyte. The electrolyte is not consumed in the process and will be reused. Larger components or those with complicated shape, which do not fit the unit, can be treated using handheld electrodes that can be fed with electrolyte.
- Wet blasting, components are decontaminated in the washing and blasting box by washing with water under high pressure and by blasting with a water and glass bead mixture.
- Chemical bating, chemical decontamination is used where radioactivity is chemically bound to the oxidised surface layer found on a component. After chemical decontamination the used decontamination chemicals are radioactive and must therefore be treated as radioactive waste.
- Manuel cleaning, by manually scrub of the surface contamination using solvents.
- High pressure washing, after all of the above methods of decontamination the parts are washed using high pressure water to remove any residues from the previous methods of decontamination.

All objects are thereafter measured for their activity, if to high they will have to be decontaminated further repeating the above procedures or reverting to other methods.

4.2 The wet blasting procedure

When a radioactive part is to be de-contaminated in the wet blaster it is first washed using a normal high-pressure water washer, as described in the above paragraph. This will remove all lose particles not bound in oxide layers. This procedure is performed in a separate enclosed compartment/cabinet and the part will then be moved to the wet blasting cabinet.

Depending on the size and geometry of the part it will either be blasted from one of the external operator positions or the operator/operators will operate the blasting equipment inside the cabinet. If the operator/operators are inside the cabinet they will be wearing special protective clothing equipped with a fresh air supply, see figure 5. Inside the cabinet there is a negative pressure to ensure no leakage of airborne particles from the cabinet. The negative pressure is obtained by a permanent connection of the cabinet to an extractor fan system. In the fan intake there is a trap that catches the bulk of glass beads and water particles present in the air. This trap is regularly flushed with water that will return the contaminated

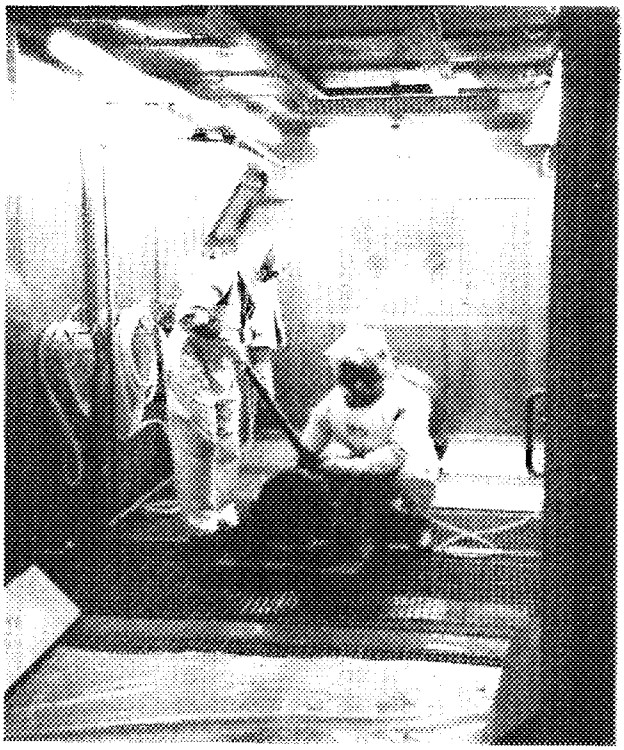

Figure 5.

water, glass beads and any lose oxide particles trapped back into the media tank.

The media tank contains the glass beads and they will be re-circulated throughout the blasting period. The beads will slowly deteriorate in the blasting process. This will lead to a diminished effectiveness in blasting off the oxide layers. However the field experience is that the beads will be replaced as a result of build-up in radioactivity rather than due to loss in blasting efficiency.

Once the operator/operators have blasted the part to desired surface finish, based on experience. The part will be washed with water to remove any glass beads stuck to the part. If the operator/operators are working inside the blasting cabinet they will shower themselves using the shower in the cabinet to remove any glass beads stuck to their protective clothing. The cabinet has a self rinsing washing system which will flush the inside of the cabinet in order to return all glass beads stuck to the ceiling, walls or floor grating to the media tank.

4.3 The waste

As previously described the waste is after a decontamination run in the blasting box trapped in either the ventilation system filter, media water or the glass beads in the media tank. How much each system has obtained will depend on the contamination of parts decontaminated and how successfully they have been cleaned.

4.3.1 The airborne waste

The pre-filter in the blasting cabinet consists of many layers of a fine metal net.

After two years of operation at OKG the build-up of activity in the filter was not sufficiently large to motivate dismantling of the filter unit.

The particles not trapped will travel trough the ventilation system and be handled in system 341, system for radioactive gaseous waste. In system 341 air borne radioactive particles from several areas including the decontamination department are collected in active carbon filters.

The total contribution of radioactivity to the carbon filter system is difficult to determine. But as the total operation time of the blaster at OKG last year was approximatley17 hours and the air-handling units are in constant operation, the operational time of the blasting equipment only constitutes 0,2 % of the operating time of the air-handling system. As the overwhelming bulk of radioactive particles are trapped and flushed back into the media tank the airborne contribution to the active carbon filters can be neglected.

4.3.2 Contamininated water treatment

Water used in the blasting system is re-circulated when blasting although there is a surplus of water added to the system when parts are being washed with "fresh" water. This water will overflow in the media tank and is pumped into one of the water treatment systems of the power plant. At BKAB this system has code 344, system for active liquid waste, the waste water will follow the path in figure 6.

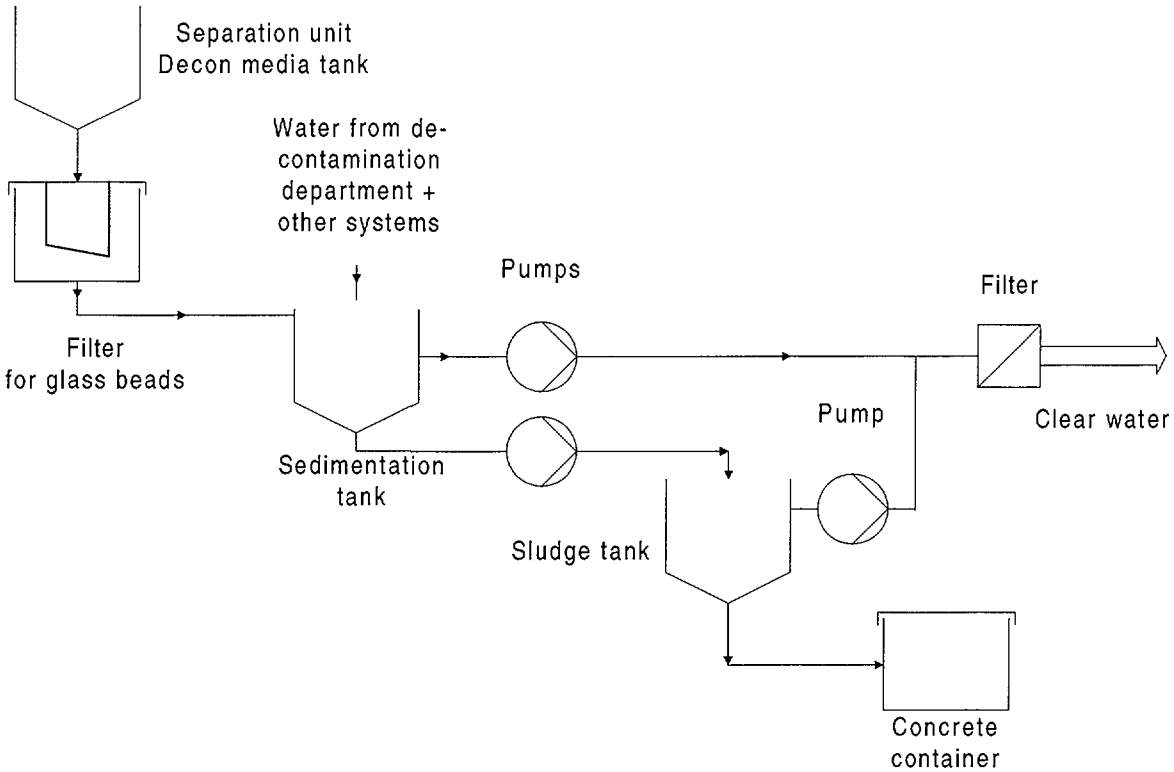

Figure 6.

From the decontamination department the water is pumped to a sedimentation tank and in this tank the solids will form sludge. This sludge is pumped into a sludge tank where it will have a second opportunity to sediment. The solids in the sludge tank will be pumped into a concrete container for waste disposal. This waste is classified as medium active waste and will be stored at SFR. After the sedimentation and sludge tank the water is pumped through a filtration unit after which it is tested for radioactivity. Once cleared it will be pumped back into the sea.

[3]The water used [6] in the wet blaster and pumped into system 344 is a result of either the part being flushed with fresh water or the media tank being emptied. The glass beads are replaced approximately 10 times per year and when this is done approximately 250 litres of water from

[3] These figures apply to BKAB and their operating procedures.

the media tank will be pumped to system 344 each time. The water used for cleaning the parts will overflow the media tank and be pumped into system 344 on a continuos basis. The total water use in the wet blaster, as a result of flushing and emptying the media tank, is estimated to 150 m3 per year. The sedimentation tank which services both system 344 and other systems holds 120 m3 and is emptied approximately 80 times per year totalling 9 600 m3 per year. This volume of water generates a sludge volume of 5,5 m3 per year. If the water from the decontamination department is as contaminated as water from the other systems, the total sludge volume contributed from the wet blaster is 150/9600 x 5,5 = 0,086 m3 or 86 litres. This would be the volume of sludge per year requiring storage at SFR as a result of the water contamination by the wet blaster. When the database is checked it turns out that 791 items has been de-contaminated in the wet blaster at BKAB over a period of one year, see appendix C1.

4.3.3 Radioactive waste mixed with the blasting material

The bulk of the radioactive oxide blasted from the treated parts is trapped in the media tank and mixed with the glass beads. When the radioactivity of the beads in the media tank has reached a certain level the beads are replaced. There are no set level at the decontamination departments at the Swedish nuclear power plants when to replace the glass beads. It is the opinion of the personnel at OKG that the level rarely exceeds 5 mSv/h before it is replaced.

At Ringhals [7] power station they check the activity of the glass beads prior to storage and if the activity is <0,5 mSv/h it will be put in a "big-bag" and taken care of by the in-house waste disposal unit If the activity level exceeds 0,5 mSv/h it will be put in a concrete container and sent to SFR.

The activity level of the glass beads is not believed to contribute to the activity level of the part being blasted i.e. the activity level of a part will not increase as a result of a higher activity level of the blasting material. The reason for this is that the activity is oxide bound and once the part is blasted the oxide is removed and the activity has a limited opportunity to stick to the surface of the part.

In order to minimise the dosage taken by the operator of the blasting equipment and to prevent build-up of activity in the various parts of the blasting system a regular change of the glass beads is of importance. At BKAB and OKG the amount of blasting material used at each site is about 400kg per year. This amount is of course dependent on the volume of goods being de-contaminated. The glass beads costs approximately 525 EURO per 100 kg.

When the glass beads are replaced they are pumped out of the media tank together with the water in the tank. The glass beads/water sludge is pumped through a cyclone filtration unit where the water and glass beads are separated. The 4beads are then pumped into a concrete storage tank and the water is pumped into system 344 and handled as described in paragraph 4.3.2. The glass beads which are contaminated by the radioactive oxide is, after the concrete tank is sealed off with a lid, transported to a storage area awaiting transportation and final storage at SFR.

4.4 Waste volumes

The various volumes of waste generated through air filtration units, water cleaning systems and used glass beads. Is dependent on the volume of goods and the contamination degree of goods handled in the blasting cabinet. It is an ambition with this report to find parameters predicting these volumes based on the surface contamination and area of a part. The chosen part to study is a shaft to a pump in system 313, the main circulation pump to the reactor tank.

[4] Applies to procedures at BKAB.

These pump shafts are regularly serviced and data from a number of these units are available. It is also one of the few items after which the glass beads are usually replaced and will therefore give an indication of contamination accepted to the blasting media before it is replaced.

5 Decontamination data from the power stations

5.1 Data from the decontamination department at OKG and BKAB

When a part is dismounted in the plant and taken to the decontamination department for treatment it is measured for activity on arrival. When measuring the kBq/m2 level they use a smear test and the reason for this test is to find out how much of the present radiation is loose radiation, these levels can vary substantially depending on the hardness of the oxide level. Dose levels are also measured in order to establish the time the service personnel can be exposed to the part.

Depending on which part it is and how the part is to be serviced or used various treatments are considered. The activity of the part has to be lowered to acceptable levels in order for the planned service to be carried out. Each part or batch of parts are measured and registered in a database. This data base provides information on wet blasting ranging back to 1994 at both BKAB and OKG Data has been extracted from the databases at OKG regarding the decontamination reports for pump shaft 313.

5.2 Data at OKG

5.2.1 313 HCP shaft

At OKG two pump shafts to the main reactor circulation pumps (HCP) where found to have been de-contaminated in 1998. They both belonged to OKG 3 the largest and newest reactor at the power plant. The following information where extracted from the OKG database:

Table 5.

Component.	System	Date	Blast time (min)	Before mSv/h	After mSv/h	Before Kbq/m2	After Kbq/m2
HCP-SHAFT	313	980120	60	78	10	>10000	300
HCP-SHAFT	313	980122	60	170	15	>10000	3300

As seen in table 5 a substantial reduction in both measured dose and activity level is achieved. This radiation is now contained in the media tank as blasted of oxide mixed with the glass-beads/water sludge.

The difference in measured dose rate depends [8] on different running time for the two pumps. Longer running time results in the forming of a thicker and harder oxide layer. The reason for the levels in table 5 being >10000 kBq/m2 is that when the levels are as high as these the radiation is measured using an instrument that indicates the level rather than give a precise reading. In reality it is more than 10000 kBq/m2, but for the purpose of determining the required protective clothing this value provides sufficient information.

Another measurement of interest to the operators is the decontamination factor. This is the relation between the dose or activity level before and after the component has been de-contaminated. The activity levels in table 5 will result in dose decontamination factors ranging from 7,8-11,3 and the activity decontamination level ranging from 33-3,3. The operators say that a decontamination factor of up to 50 is achievable using wet blasting. To reach a decontamination factor of 50 on the above shafts the blasting time would have had to be substantially increased.

5.2.2 Activity contribution to blasting media

The area of the shaft which has been blasted is approximately 0,386 m2 and the level of reduced radiation is, on the shaft de-contaminated 980120, >9700 kBq/m2. The media tank now contains the blasted of oxide layers in excess of 0,386 x 9700 = 3744 kBq. The media tank contains approx. 40 kg of glass beads, assuming a homogenous distribution of the blasted off oxide, the contamination in the tank is after the cleaning of the shaft in excess of 93 kBq/kg. The removed level of activity is probably less as the measurement on the shaft is the peak value and the average value is most likely to be less.

The level of radiation, if acceptable for ground level storage, has to be less than 300 kBq/kg. This should mean that the glass beads from a decontamination run on a 313 HCP-shaft leaves the glass beads less contaminated, even if some other parts have been de-contaminated using the same glass beads. If the glass beads are changed, as indicated at OKG, after a decontamination run on one 313 HCP-shaft, the radioactive glass beads should not require storage at SFR. Various storage types carries different costs and if a ground storage facility can be used, they normally are on the grounds of the power plant, savings can be made from both transportation and storage.

At OKG the contaminated glass beads are mixed with ion exchanging material from the cleaning of decontamination chemicals, this material only constitutes approx. 10%. This mixture has to be deposited at SFR.

6 Declassification of contaminated parts

6.1 Declassification and wet blasting

Under normal conditions wet blasting is used to remove oxide layers on metal. When these oxide layers forms in the "hot" systems around the plant experience has proven it difficult to achieve the low levels of activity required for the part to be de-classified. The reasons being that these systems are made out of high quality materials with a fine surface, which still has a lot of very fine cavities. To lower the activity level is regarded as easy but to get the dosage level down can be very difficult. The reason for this comes from the way the radiation is measured. The activity level is measured using a smear test where contamination is rubbed of against a sample and then the sample is measured. The dose level is measured using an instrument, which detects the radiation from the surface. What the dose meter reads is the radiation trapped in the cavities of the material and this radiation is very difficult to remove.

A search of the decontamination databases at BKAB and OKG to see how many parts have been declassified using wet blasting gives the result found in table 7.

Table 7.

Site	Total number of entries in the database.	Number Declassified	Number Blasted and declassified	Prior to treatment max. dose/activity, mSv/kBq/m^2
BKAB	1758	301	28	42/30000
OKG	1122	39	9	20/>10000

Very few of the total number of entries have been declassified as a result of wet blasting. One of the reasons for this is that the operators have no interest to reach such a low level of contamination. Their ambition is to remove the bulk of radiation to such a level where they have optimised their taken dose in relation to the overall dose of the service personnel to be subjected to the specific part.

When a part needs to be declassified, which they have to if they are to be taken out of and used outside zone controlled areas. The part normally goes through a variety of treatments, as previously described, and no one treatment can be singled out as the "de-classifier". However there are two main methods of reducing the activity level and they are wet blasting and electropolishing. Electropolishing has the disadvantage [9] of contributing more dose to the operator due to the exposure when fitting the electrodes. The operator is in close contact with the part which is not the case when wet blasting where the operator/operators are at some distance.

6.2 Lowering the radiation level to allow melting

Studsvik™ RadWaste AB is specialised in melting radioactive metal, the metal melting facility is situated at Studsvik 100 km south of Stockholm

Studsvik™ RadWaste offers melting of active metal. This is a way to reduce the volume of material as well as the activity. The cost for handling and melting the metal parts is approximately 2.30 EURO/kg.

The following requirements apply to materials sent for melting.

Delivery specifications:

Metals as: stainless steel, carbon steel, aluminium, copper and brass.

No residuals of liquids, toxic and organic.

Surface dose rate: < 0.5 mSv/h

Melting has the advantage of lowering the volume of goods to be stored and it also lowers the activity level of the goods. The activity level is lowered by some of the activity vaporising allowing it to be trapped in filters some being removed in the slag and also by evening out the mainly surface bound activity in the full volume of the material.

The most common nuclides found in the ingots, the nuclides half times and the activity content as a result of the melting is presented in table 8.

Table 8. [5]

Nuclide	**$T_{1/2}$ Years**	**Activity contents in %**		
		Ingot	**Slag**	**Dust**
Mn-54	0.9	84	15.8	0.2
Co-60	5.3	99.6	0.3	0.07
Zn-65	0.7	90.1	0.5	9.4
Ag-110m	0.7	0	0	100
Cs-134	2.2	0	79.9	21.0
Sb-125	2.7	0	0	100
Cs-137	30	0	73	27

We find that some of the above nuclides are completely removed from the ingot as a result of the melting. However the main nuclide being Cobalt 60 largely remains. It is also Cobalt 60 that has the longest half time of the remaining nuclides with 5.3 years.

The ingots will be stored and monitored and if their activity level falls below the limit value they will be recycled in the metal industry. Slag, filter dust, out sorted material is sent back to the power plant in drums or containers.

Radiological specifications for releasing of ingots for unrestricted use:

Alpha-nuclides: < 100 Bq/kg

Gamma/Beta-nuclides: < 1000 Bq/kg

However if the surface contamination exceeds 15-16 Bq/gram [5] metal and this activity mainly derives from the nuclide cobalt 60, as is common in nuclear power plants, the ingot will not be recyclable in 20 years. If the activity level after 20 years still exceeds the acceptable level the ingot will be returned to the power plant that sent the metal waste to Studsvik™ Rad Waste. If returned the metal will end up having to be stored at SFR with its additional costs.

If parts are de-contaminated at the power plants before being sent to SFR the existing blasting equipment at the power station can be utilised. This would enable the operators of the power plant insure that the level of activity do not exceed 15 Bq/gram metal. Thus not risking having the ingots sent back after the 20 years of decay time.

7 The economics of treating components using glass blasting

7.1 Current costs for radioactive waste storage

As mentioned in paragraph 3, radioactive waste is divided into different categories. When it comes to waste related to wet blasting the waste activity will be of low and medium level. When the Swedish nuclear program was decided upon the waste storage facility at Forsmark was built in order to accommodate low and medium level radioactive waste. This facility was built and paid for as an integrated part of the whole nuclear program and each power station has part of the storage space. The storage space consists of one part for low level waste and one unit for medium level waste.

Low-level waste is stored in standard containers in large rock vaults. Because the waste emits such low levels of radiation they can be handled using normal forklift trucks.

Medium level waste is stored and handled in special concrete containers each with a total volume of 1.44 m3 of which 0.3 m3 is available for the waste. The concrete containers are remotely handled but do not require cooling.

The storage costs will have to be estimated on what it would cost to build new facilities to store the waste. The existing facilities will in time be full and the waste that can be decommissioned today will save a future cost.

Table 9. [6][8][9]

	Low level waste stored at ground level[1]	Low level waste	Medium level waste
Storage costs, EURO/m^3	580	986	19 376

[1]As are done at OKG.

7.2 Possible savings through wet blasting

We need to remember that the primary objective of wet blasting is to minimise the dose to service personnel at the power stations. This is as described achieved through decontamination of parts thus facilitating necessary service to components which otherwise would be highly radioactive.

The wet blasting equipment is however at hand at the power stations and they could systematically use it in order to ensure that goods being sent for melting have an activity,

which allows the ingots to be recycled within the 20-year period. As the power stations age the volume of goods being sent for melting will increase.

The wet blaster could also, without any drastic changes, be converted to use a more abrasive blasting media such as aluminium silica. This would allow parts to be de-contaminated to declassification levels thus allowing the part to be recycled using ordinary channels. By increasing the effectiveness of the media the required blasting time would decrease. Shortening the exposure time per item to the operator. The dos taken by the operators of the blasting equipment and contributed to the blaster is today in the order of 1 mSv per year. In appendix C1 the number of parts handled, at BKAB, in one year was found to be 791. This indicates the possibility that a large number of items could be de-classified this way without adding a substantial amount of dose to the operators. This way of declassifying parts requires that the dose taken when handling the components in and out of the wet blaster is under control and not excessively add to the total dose taken by the operator.

The savings will come from lower storage costs as well as a saving on the environmental impact from the benefit of the metal being recycled rather than deposited. In the case of using a more abrasive blasting material savings would also be made on the cost of melting metal components at Studsvik™ Rad Waste.

7.3 Waste costs derived from the wet blaster at BKAB

At BKAB nearly 800 components where decontaminated using wet blasting from 980501-990531. At the same time our estimates shows that 86 litres of sludge and 400 kg of blasting material will have to be taken care of as a result of the decontamination of these components. The 400 kilos of blasting material equals approximately 160 litres in volume and is divided up in 10 off 200 litres concrete lined barrels totalling 2 m3 of required storage space. The sludge from the filtration of the water will have to be stored at SFR's medium level storage facility at a cost of 0,086 x 19 376= 1 666 EURO. While the used glass beads will have to be stored at the low level waste facility at SFR costing 2 x 986 = 1 972 EURO. The total storage cost for the derived waste products from the wet blasting equipment at BKAB for one year of operation is then 3 638 EURO.

8 Conclusions

The wet blasting equipment at the two nuclear power stations in focus for this report has proven a capability of substantially lowering the surface contamination on contaminated components. Having studied the wet blasting technique one finds it to be an indispensable part of a modern well functioning decontamination department.

There are drawbacks and the most prominent is that the operators still are exposed to radiation when blasting bigger components thus having to physically be in the cabinet. This is however a problem possible to solve by using a blasting robot, which could be controlled from one of the external operators positions where the operator benefits from radiation protection built into the cabinet.

If wet blasting is compared to electropolishing, the other option of oxide removing decontamination treatment. Wet blasting has the advantage of providing a comparable decontamination factor exposing the operator to less dose and leaving derived waste products easier to handle. Wet blasting is, compared to other methods of decontamination, a fast way to decontaminate components, which are radioactive allowing a great number of components to be treated in a short time space. The blasting cabinet at BKAB and OKG, offers the option to decontaminate parts prior to melting at Studsvik™ thereby ensuring the activity levels

required to allow the metal ingots to be recycled within the 20 year time span provided. If the cabinet was altered to enable it to use a more abrasive blasting material many components could be treated prior to being scrapped. Either declassifying them or at least meeting the requirements for melting resulting in considerable savings when a plant is to be decommissioned as is the case at BKAB.

At BKAB nearly 800 components where de-contaminated in the wet blaster in the past year and this resulted in low and medium level waste with a storage cost of approximately 3 640 EURO. After the components had been blasted they all reached a level of activity, which allowed the components to be handled without contributing excessively to the dose taken by the service personnel.

The study of international organisations recommended/legislated levels for clearance showed a variation ranging from 150 Bq/kg to 1000 Bq/kg of Beta and Gamma radiators. This variation indicates that the Swedish levels for clearance are placed in-between at 500 Bq/kg of Beta and Gamma radiating nuclides.

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Appendix A: Governing organizations

Table1. Derived levels for clearance in accordance with IAEA [11].

Interval of specific activity (Bq/kg)	Radio nuclide	Representative value (Bq/kg)
10^2 $<10^3$	Na-22, Na-24, *Mn-54*, *Co-60*, *Zn-65* Nb-94, *Ag-110m*, Sb-124, *Cs-134*, *Cs-137*, Eu-152, Pb-210, Ra-226, Ra-228, Th-228, Th-230, Th-232, U-234, U-235, U-238, Np-237, Pu-239, Am-241, Cm-244	$3*10^2$
$\geq 10^3$ $<10^4$	*Co-58*, Fe-59, Sr-90, *Ru-106*, In-111, I-131, Ir-192, Au-198, Po-210	$3*10^3$
$\geq 10^4$ $<10^5$	*Cr-51*, Co-57, Tc-99m, I-123, I-125, I-129, *Ce-144*, Tl-201, Pu-241	$3*10^4$
$\geq 10^5$ $<10^6$	C-14, P-32, Cl-36, *Fe-55*, Sr-89, Y-90, Tc-99, Cd-109	$3*10^5$
$\geq 10^6$ $<10^7$	H-3, S-35, Ca-45, *Ni-63*, Pm-147	$3*10^6$

Note: Nuclides in cursive stile are commonly found in waste from nuclear power plants.

Table 2. Derived levels for direct reuse of equipment and tools according to OECD/NEA [13].

Nuclide category	Representative nuclides	Specific activity (Bq/kg)	Surface contamination (kBq/m^2)
Alpha radiator	Ac-227 Am-241 Pu-239 U-238	$1,1*10^3$ $4,6*10^3$ $6,9*10^6$ $1,6*10^4$	12 110 150 370
Beta radiator	Sr-90 Tc-99	$9,4*10^4$ $1,6*10^7$	2700 $1,7*10^5$
Gamma radiator	Co-60 Cs-137 Zn-65	$1,5*10^2$ $6,7*10^2$ $9,8*10^2$	11 44 68

Table 3. Derived levels for direct reuse of equipment and tools (steel) according to the EU-commission [14].

Nuclide	Surface contamination (kBq/m^2)		Nuclide	Surface contamination (kBq/m^2)
H-3	100000		Cs-137	100
C-14	10000		Pm-147	10000
Mn-54	100		Sm151	10000
Fe-55	10000		Eu-152	1
Co-60	10		Eu-154	1
Ni-59	100000		U-234	1
Ni-63	10000		U-235	1
Zn-65	100		U-238	1

Sr-90	10		Np-237	1
Nb-94	10		Pu-238	1
Tc-94	10000		Pu-239	1
Ru-106	100		Pu-240	1
Ag-108m	10		Pu-241	10
Ag-110m	10		Am-241	1
Sb-125	100		Cm-244	1
Cs-134	10			

Table 4. Scenarios for recycling of steel: dosage interval (μSv/year and 1 kBq/kg in the metal waste) and the predominant way of exposure [12].

Nuclide	Dosage interval (dominant way of exposure)	
	Steel mill	Consumer product
Co-60	1 – 10 (ext)	10 – 100 (ext)
Sr-90	0,2 – 2 (inh)	< 0,1 (int)
Cs-70	0,7 – 5 (ext)	< 0,01 – 20 (ext)
Pu-239	1 – 1000 (inh)	< 0,001 – 2 (int)
Am-241	1 – 1000 (inh)	< 0,001 – 5 (int)

ext = external radiation, inh = inhalation, int = intake,

Table 5. Comparing EU's [14] and OECD/NEA's [13] radiation levels for melting and recycling of metal (normalized to 10000 tons per year).

Nuclide category	Representative nuclide	EU [7]		OECD/NEA [6]	
		Limiting boundary scenario	Specific activity[a] (Bq/kg)	Limiting boundary scenario	Specific activity[a] (Bq/kg)
Alpha	Am-241	Field with slag, football player	$1*10^3$	Slag worker	$2*10^4$
	Pu-239	Field with slag, football player	$1*10^3$	Slag worker	$3*10^4$
	U-238	Field with slag, football player	$1*10^3$	Slag worker	$2*10^4$
Beta	Sr-90	Large object	$1*10^4$	Slag worker	$1*10^5$
	Tc-99	Slag, people living on landfill site	$2*10^5$	Slag worker	$1*10^6$
Gamma	Co-60	Worker on boat	$1*10^3$		$3*10^3$
	Cs-137	Dust, worker on landfill site	$1*10^3$	Slag worker	$1*10^3$
	Zn-65	Dust, worker on landfill site	$1*10^3$	Slag worker	$1*10^5$

a. Equivalent to a individual dos of 10 μSv/year

Table 6. Proposal for clearance levels for copper and aluminium [15].

Nuclide	Copper		Aluminium	
	Clearance level (Bq/year)	Max. Individual dosage (μSv/year)	Clearance level (Bq/year)	Max. Individual dosage (μSv/year)
H-3	$7.7*10^{7}$	$1,3*10^{-4}$	$1,6*10^{7}$	$6,2*10^{-4}$
C-14	$3,1*10^{6}$	$3,2*10^{-3}$	$6,4*10^{5}$	$1,6*10^{-2}$
Mn-54	$1,7*10^{4}$	0,6	$3,2*10^{3}$	3,2
Fe-55	$4,6*10^{7}$	$2,2*10^{-4}$	$9,8*10^{7}$	$1,0*10^{-4}$
Co-60	$5,3*10^{3}$	1,9	$1,0*10^{3}$	11,0
Ni-59	$1,2*10^{7}$	$8,6*10^{-4}$	$9,6*10^{7}$	$1,0*10^{-4}$
Ni-63	$3,9*10^{6}$	$2,6*10^{-3}$	$3,2*10^{7}$	$3,1*10^{-4}$
Zn-65	$2,4*10^{4}$	0,4	$5,5*10^{3}$	1,8
Sr-90+	$8,9*10^{3}$	1,1	$4,8*10^{3}$	2,1
Nb-94	$1,3*10^{3}$	7,9	$1,0*10^{3}$	22,0
Tc-99	$1,4*10^{6}$	$6,9*10^{-3}$	$2,0*10^{6}$	$15,0*10^{-3}$
Ru-106+	$1,5*10^{4}$	0,6	$3,3*10^{3}$	3,0
Ag-108m	$1,0*10^{3}$	9,5	$1,0*10^{3}$	11,0
Ag-110m	$1,2*10^{3}$	8,0	$1,1*10^{3}$	9,4
Sb-125+	$4,8*10^{3}$	2,1	$1,0*10^{3}$	5,3
Cs-134	$9,0*10^{3}$	1,1	$1,0*10^{3}$	21,0
Cs-137+	$2,5*10^{4}$	0,4	$1,3*10^{3}$	7,6
Pm-147	$8,4*10^{7}$	$1,2*10^{-4}$	$1,0*10^{7}$	$9,5*10^{-4}$
Sm-151	$1,3*10^{8}$	$7,7*10^{-5}$	$2,1*10^{7}$	$4,7*10^{-4}$
Eu-152	$2,9*10^{3}$	3,5	$1,0*10^{3}$	19,0
Eu-154	$3,6*10^{3}$	2,8	$1,0*10^{3}$	11,0
U-234	$1,4*10^{3}$	7,2	$3,2*10^{3}$	3,2
U-235+	$1,4*10^{3}$	7,2	$3,2*10^{3}$	3,2
U-238+	$1,4*10^{3}$	7,2	$3,2*10^{3}$	3,2
Np-237+	$1,0*10^{3}$	14,4	$1,6*10^{3}$	6,3
Pu-238	$1,0*10^{3}$	14,4	$1,6*10^{3}$	6,3
Pu-239+	$1,0*10^{3}$	14,4	$1,6*10^{3}$	6,3
Pu-240	$1,0*10^{3}$	14,4	$1,6*10^{3}$	6,3
Pu-241+	$4,6*10^{4}$	0,2	$4,61,0*10^{5}$	$9,5*10^{-2}$
Am-241	$1,0*10^{3}$	14,4	$1,6*10^{3}$	6,3
Cm-244	$1,2*10^{3}$	8,6	$2,6*10^{3}$	3,8

Note. Values based on 200 ton of copper per year and 40 ton of aluminium per year (this is equivalent to the amount in two PWR-reactors)

Table 7. Clearance levels based on 10 μSv/year to individuals in a critical group [10].

Nuclide	Clearance level (Bq/kg)	Predominate way of exposure
C-14	$9,1*10^{5}$	Food intake
Na-22	$1,8*10^{7}$	External exposure
P-32	$1,0*10$	Food intake and inhalation
S-35	$3,1*10^{7}$	Food intake
Ca-45	$1,8*10^{7}$	Food intake and inhalation
Mn-54	$1,3*10^{9}$	Food intake
Co-60	$7,7*10^{7}$	Food intake and inhalation
Sr-90	$7,7*10^{7}$	External exposure
Ru-106	$7,1*10^{7}$	External exposure
I-131	$3,7*10^{6}$	Food intake and inhalation

Cs-137	$1,3*10^7$	External exposure
Ce-144	$1,0*10^9$	Food intake and inhalation
Pu-239	$7,1*10^5$	Inhalation
Am-241	$7,1*10^5$	Inhalation

In Finland [7] the following levels for declassification and clearance applies.
Specific activity:

< 1 kBq/kg Beta and Gamma radiation
< 0,1 kBq/kg Alpha radiation

No singular object may contain more than 100 kBq Beta or Gamma radiating nuclides or 10kBq Alpha radiating nuclides.
Non-fixed surface contamination may not exceed:

< 4 kBq/m^2 Beta and gamma radiation
< 0,4 kBq/m^2 Alpha radiation

Appendix B: Specific radio nuclide guidelines, EU

The guidelines of the European Community Radiation Protection Legislation document X1-3539/96-EN says in Article 5 and Annex 1:

1. The disposal, recycling or reuse of radioactive substances or materials containing radioactive substances arising from any practice subject to the requirement of reporting or authorization is subject to prior authorization.
2. However, the disposal, recycling or reuse of such substances or materials may be released from the requirements of this Directive provided they comply with clearance levels established by national competent authorities. These clearance levels shall follow the basic criteria used in Annex 1 and shall take into account any other technical guidance provided by the Community.

ANNEX 1

1. A practice may be exempted from the requirement to report without further consideration in compliance with Article 3 (2) or (b) respectively, if either the quantity or the activity concentration, as appropriate, of the relevant radio nuclides does not exceed the value in column 2 or 3 of Table A
2. The basic criteria for the calculation of the values in Table A, for the application of exemption for practices, are as follows:

   (a) the radiological risks to individuals caused by the exempted practice are sufficiently low as to be of no regulatory concern; and

   (b) the collective radiological impact of the exempted practice is sufficiently low as to be of no regulatory concern under the prevailing circumstances; and

   (c) the exempted practice is inherently without radiological significance, with no appreciable likelihood of scenarios that could lead to a failure to meet the criteria in (a) and(b).

3. Exceptionally individual Member States may decide that a practice may be exempted where appropriate without further consideration, in accordance with the basic criteria, even if the relevant radio nuclides deviate from the values in Table A, provided that the following criteria are met in all feasible circumstances:

   (a) the effective dose expected to be incurred by any member of the public due to the exempted practice is of the order 10 μSv or less in a year;

   and

   (b) either the collective effective dose committed during one year of performance of the practice is no more than about 1 man x Sv or an assessment of the optimization of protection shows that exemption is the optimum option

4. For radionuclides not listed in Table A, the competent authority shall assign appropriate values for the quantities and concentration of activity per unit mass where

the need arises. Values thus assigned shall be complementary to those in table A.

5. The values laid down in Table A apply to the total inventory of radioactive substances held by a person or undertaking as part of specific practice at any point in time.

6. Nuclides carrying the suffix `+` or `sec` in Table A represent parent nuclides in equilibrium with their correspondent daughter nuclides as listed in Table B. In this case the values given in Table A refer to the parent nuclide alone, but already take account of the daughter nuclide(s) present.

7. In all other cases of mixtures of more than one nuclide, the requirement for reporting may be waived if the sum of the ratios for each nuclide of the total amount present divided by the value listed in Table A is less than or equal to 1. This summation rule also applies to activity concentrations where the various nuclides concerned are contained in the same matrix.

TABLE A

Nuclide	Quantity (Bq)	Concentration (kBq/kg)		Nuclide	Quantity (Bq)	Concentration (kBq/kg)
H-3	10^9	10^6		Zn-69	10^6	10^4
Be-7	10^7	10^3		Zn-69m	10^6	10^2
C-14	10^7	10^4		Ga-72	10^5	10
O-15	10^9	10^2		Ge-71	10^8	10^4
F-18	10^6	10		As-73	10^7	10^3
Na-22	10^6	10		As-74	10^6	10
Na-24	10^5	10		As-76	10^5	10^2
Si-31	10^6	10^3		As-77	10^6	10^3
P-32	10^5	10^3		Se-75	10^6	10^2
P-33	10^8	10^5		Br-82	10^6	10
S-35	10^8	10^5		Kr-74	10^9	10^2
Cl-36	10^6	10^4		Kr-76	10^9	10^2
Cl-38	10^5	10		Kr-77	10^9	10^2
Ar-37	10^8	10^6		Kr-79	10^5	10^3
Ar-41	10^9	10^2		Kr-81	10^7	10^4
K-40	10^6	10^2		Kr-83m	10^{12}	10^5
K-42	10^6	10^2		Kr-85	10^4	10^5
K-43	10^7	10		Kr-85m	10^{10}	10^3
Ca-45	10^6	10^4		Kr-87	10^9	10^2
Ca-47	10^6	10		Kr-88	10^9	10^2
Sc-26	10^6	10		Rb-86	10^5	10^2
Sc-47	10^5	10^2		Sr-85	10^6	10^2
Sc-48	10^5	10		Sr-85m	10^7	10^2
V-48	10^7	10		Sr-87m	10^6	10^2
Cr-51	10^5	10^3		Sr-98	10^6	10^3
Mn-51	10^5	10		Sr-90+	10^4	10^2
Mn-52	10^5	10		Sr-91	10^5	10
Mn-52m	10^9	10		Sr-92	10^6	10
Mn-53	10^6	10^4		Y-90	10^5	10^3

Mn-54	10^5	10		Y-91	10^6	10^3
Mn-56	10^6	10		Y-91m	10^6	10^2
Fe-52	10^6	10		Y-92	10^5	10^2
Fe-55	10^6	10^4		Y-93	10^5	10^2
Fe-59	10^6	10		Zr-93+	10^7	10^3
Co-55	10^5	10		Zr-95	10^6	10
Co-56	10^6	10		Zr-97+	10^5	10
Co-57	10^6	10^2		Nb-93m	10^7	10^4
Co-58	10^7	10		Nb-94	10^6	10
Co-58m	10^5	10^4		Nb-95	10^6	10
Co-60	10^5	10		Nb-97	10^6	10
Co-60m	10^6	10^3		Nb-98	10^5	10
Co-61	10^6	10^2		Mo-90	10^6	10
Co-62m	10^5	10		Mo-93	10^8	10^3
Ni-59	10^8	10^4		Mo-99	10^6	10^2
Ni-63	10^8	10^5		Mo-101	10^6	10
Ni-65	10^6	10		Tc-96	10^6	10
Cu-64	10^6	10^2		Tc-96m	10^7	10^3
Zn-65	10^6	10		Tc-97	10^8	10^3

Nuclide	Quantity (Bq)	Concentration (kBq/kg)		Nuclide	Quantity (Bq)	Concentration (kBq/kg)
Tc-97m	10^7	10^3		Xe-135	10^{10}	10^9
Tc-99	10^7	10^4		Cs-129	10^5	10^7
Tc-99m	10^7	10^2		Cs-131	10^6	10^7
Ru-97	10^7	10^2		Cs-132	10^5	10^9
Ru-103	10^6	10^2		Cs-134m	10^5	10^6
Ru-105	10^6	10		Cs-134	10^4	10^6
Ru-106+	10^5	10^2		Cs-135	10^7	10^5
Rh-103m	10^8	10^4		Cs-136	10^5	10^6
Rh-105	10^7	10^2		Cs-137+	10^4	10^5
Pd-103	10^8	10^3		Cs-138	10^4	10^8
Pd-109	10^6	10^3		Ba-131	10^6	10^8
Ag-105	10^6	10^2		Ba-140+	10^5	10^6
Ag-108m+	10^6	10		La-140	10^5	10^5
Ag-110m	10^6	10^3		Ce-139	10^6	10^8
Ag-111	10^6	10^4		Ce-141	10^7	10^9
Cd-109	10^6	10^2		Ce-143	10^6	10^6
Cd-115	10^6	10^3		Ce-144+	10^5	10^6
Cd-115m	10^6	10^2		Pr-142	10^7	10^7
In-111	10^6	10^2		Pr-143	10^6	10^6
In-113m	10^6	10^2		Nd-147	10^6	10^6
In-114m	10^6	10^2		Nd-149	10^6	10^6
In-115m	10^6	10^3		Pm-147	10^7	10^5
Sn-113	10^7	10^2		Pm-149	10^6	10^5
Sn125	10^5	10^2		Sm-151	10^8	10^7
Sb-122	10^4	10^2		Sm-153	10^6	10^5
Sb-124	10^6	10		Eu-152	10^6	10^5
Sb-125	10^6	10^2		Eu-152m	10^6	10^5
				Eu-154	10^6	10^9

Te-123m	10^7	10^2		Eu-155	10^7	10^6
Te-125m	10^7	10^3		Gd-153	10^7	10^5
Te-127	10^6	10^3		Gd-159	10^6	10^6
Te-127m	10^7	10^3		Tb-160	10^6	10^6
Te-129	10^6	10^2		Dy-165	10^6	10^6
Te-129m	10^6	10^3		Dy-166	10^6	10^6
Te-131	10^5	10^2		Ho-166	10^5	10^5
Te-131m	10^6	10		Er-169	10^7	10^6
Te-132	10^7	10^2		Er-171	10^6	10^6
Te-133	10^5	10		Tm-170	10^6	10^7
Te-133m	10^5	10		Tm-171	10^8	10^5
Te-134	10^6	10		Yb-175	10^7	10^5
I-123	10^7	10^2		Lu-177	10^7	10^6
I-125	10^6	10^3		Hf-181	10^6	10^6
I-126	10^6	10^2		Ta-182	10^4	10^5
I-129	10^5	10^2		W-181	10^7	10^8
I-130	10^6	10		W-185	10^7	10^8
I-131	10^6	10^2		W-187	10^6	10^6
I-132	10^5	10		Re-186	10^6	10^6
I-133	10^6	10		Re-188	10^5	10^6
I-134	10^5	10		Os-185	10^6	10^8
I-135	10^6	10		Os-191	10^7	10^6
Xe-131m	10^4	10^4		Os-191m	10^7	10^6
Xe133	10^4	10^3				

Nuclide	Quantity (Bq)	Concentration (kBq/kg)		Nuclide	Quantity (Bq)	Concentration (kBq/kg)
Os-193	10^6	10^2		U-231	10^4	10^2
Ir-190	10^6	10		U-232+	10^3	1
Ir-192	10^4	10		U-233	10^4	10
Ir-194	10^5	10^2		U-234	10^4	10
Pt-191	10^6	10^2		U-235+	10^4	10
Pt-193m	10^7	10^3		U-236	10^4	10
t-197	10^6	10^3		U-237	10^6	10^2
Pt-197m	10^6	10^2		U-238+	10^4	10
Au-198	10^6	10^2		U-238sec	10^3	1
Au-199	10^6	10^2		U-239	10^6	10^2
Hg-197	10^7	10^2		U-240	10^4	10^3
Hg-197m	10^6	10^2		U-240+	10^6	10
Hg-203	10^5	10^2		Np-237+	10^8	1
Tl-200	10^6	10		Np-239	10^7	10^2
Tl-201	10^6	10^2		Np-240	10^6	10
Tl-202	10^6	10^2		Pu-234	10^7	10^2
Tl-204	10^4	10^4		Pu-235	10^7	10^2
Pb-203	10^6	10^2		Pu-236	10^4	10
Pb-210+	10^4	10		Pu-237	10^7	10^3
Pb-212+	10^5	10		Pu-238	10^4	1
Bi-206	10^5	10		Pu-239	10^4	1
Bi-207	10^6	10		Pu-240	10^3	1
Bi-212+	10^6	10^3		Pu-241	10^5	10^2
Po-203	10^5	10		Pu-242	10^4	1
Po-205	10^6	10		Pu-243	10^7	10^3
Po-207	10^6	10		Pu-244	10^4	1
Po-210	10^4	10		Am-241	10^4	1
At-211	10^7	10^3		Am-242	10^6	10^3
Rn-220+	10^7	10^4		Am-242m+	10^4	1
Rn-222+	10^8	10		Am-243+	10^3	1
Ra-223+	10^5	10^2		Cm-242	10^5	10^2
Ra-224+	10^5	10		Cm-243	10^4	1
Ra-225	10^5	10^2		Cm-244	10^4	10
Ra-226+	10^4	10		Cm-245	10^3	1
Ra-227	10^6	10^2		Cm-246	10^3	1
Ra-228+	10^5	10		Cm-247	10^4	1
Ac-228	10^6	10		Cm-248	10^3	1
Th-226+	10^7	10^3		Bk-249	10^6	10^3
Th-227	10^4	10		Cf-246	10^6	10^3
Th-228+	10^4	1		Cf-248	10^4	10
Th-229+	10^3	1		Cf-249	10^3	1
Th-230	10^4	1		Cf-250	10^4	10
Th-231	10^7	10^3		Cf-251	10^3	1
Th-232sec	10^3	1		Cf-252	10^4	10
Th-234+	10^5	10^3		Cf-253	10^5	10^2
Pa-230	10^6	10		Cf-254	10^3	1
Pa-231	10^3	1		Es-253	10^5	10^2
Pa-233	10^7	10^2		Es-254	10^4	10
U-230+	10^5	10		Es-254m	10^6	10^2
				Fm-254	10^7	10^4
				Fm-255	10^6	10^3

TABLE B

List of nuclides in secular equilibrium as referred to in point 6 of this Annex

Parent nuclide	Daughter nuclides
Sr-80+	Rb-80
Sr-90+	Y-90
Zr-93+	Nb-93m
Zr-97+	Nb-97
Ru-106+	Rh-106
Ag-108m+	Ag-108
Cs-137+	Ba-137
Ba-140+	La-140
Ce-134+	La-134
Ce-144+	Pr-144
Pb-210+	Bi-210, Po-210
Pb-212+	Bi-212, Tl-208, Po-212
Bi-212+	Tl-208, Po-212
Rn-220+	Po-216
Rn-222+	Po-218, Pb-214, Bi-214, Po-214
Ra-223+	Rn-219, Po-215, Pb-211, Bi-211, Tl-207
Ra-224+	Rn-220, Po-216, Pb-212, Bi-212, Tl-208, Po-212
Ra226+	Rn-222, Po-218, Pb-214, Bi-210, Po-210, Po-214
Ra-228+	Ac-228
Th-226+	Ra-222, Rn-218, Po-214
Th-228+	Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208, Po-212
Th-229+	Ra-225, Ac-225, Fr-221, At-217, Bi-213, Po-213, Pb-209
Th-232sec	Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208, Po-212
Th-234+	Pa-234m
U-230+	Th-226, Ra-222, Rn-218, Po-214
U-232+	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208, Po-212
U-235+	Th-231
U-238+	Th-234, Pa234m
U-238sec	Th-234, Pa-234m, U-234, Th-230, Ra-226, Rn-222, Po-218, Pb-214, Bi-214, Pb-210, Bi-210, Po-210, Po-214
U-240+	Np-233
Np-237+	Pa-233
Am-242+	Am-242
Am-243+	Np-239

Appendix C: Data from BKAB

Excerpts from different runs of the data bases of the decontamination records.

C1shows the all the parts de-contaminated using wet blasting at BKAB for the period of one year.

Part	No.	Date	Dosr before mSv/h	Dosr after mSv/h	Surface before Bq/kvcm	Surface after Bq/kvcm	Blasting	De-classification	Ellectropolishing
VALVE	1	9905	15	2.5	4250	4	X		
WEDGE	1	9905	60	9	1000	10	X		
VALVE	3	9905			10	2	X		
VALVE PARTS	2	9905	0.05	0.02	200	2	X		
WEDGE	1	9905	6.5	2	1000	4	X		
PUMP PARTS	2	9905			10	2	X		
PUMP PARTS	2	9905			10	2	X		
CRDM PARTS	500	9812	1.5		2000	2	X		X
WELDING EQUIP.		9811			12	0.3	X	X	
LATHES	3	9811	0.06		20	0.3	X	X	
INDUCER	1	9811			30	1	X	X	
SHAFT	1	9811	12	0.6			X		
IMPELLER	1	9811	40	2			X		
CUTTING DEVICE	250	9811	1		1000	0.3	X	X	
IMPELLER	1	9810	65	8	4000	4	X		
PUMP PARTS	4	9810			2000	20	X		
IMPELLER	1	9809	18	4	4000	3	X		
HEAT SHIELD	1	9809	14	4	3000	10	X		
HOUSING	1	9809	20	1	3000	20	X		
STEM	1	9809	40	1	3000	20	X		
WEDGE	1	9809	80	20	3000	20	X		
PULL ROD	2	9809	100	0.5	3000	20	X		
WEDGE	1	9809	50	10	3000	20	X		
STEM	1	9809	25	4	3000	20	X		
HOUSING	1	9809	22	1	3000	20	X		
PULL ROD	2	9809	100	0.5	3000	20	X		
VALVE PARTS	1	9809			10	2	X		
PUMP PARTS	2	9807	0.01	0.01	16	2	X		
SPRINGS	3	9805			5	2	X	X	

From 9805 to 9905 a total of 791 items has been de-contaminated.