行政院原子能委員會放射性物料管理局
委託研究計畫研究報告

# 除役廢棄物分類包裝審查技術研究

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# Research on Review of Packages and Classification of Decommissioning Waste

By

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Abstract

Large amounts of low level radioactive waste will be generated during the decommissioning of a nuclear power plant. All of the decommissioning waste should be controlled and safely managed for dry storage, transportation, or final disposal, as appropriate. This report would preliminary study on the related regulatory of low level wast packages and the guidances for the review approval of application for packages. The results of the report could provide experience of review approval of application for packages used to decommissioning waste.

Keywords: Decommission, Low-level radioactive waste, Review of packing

Institute of Nuclear Energy Research

# 除役廢棄物分類包裝審查技術研究

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## 摘　要

核能電廠除役期間將會產生龐大各類型放射性廢棄物，這些廢棄物經過分類及除污程序後，部分須在最終處置前加以處理與安定化，由不同效能之盛裝容器密封貯存。由於國內核能電廠即將面臨除役之需求，需要建立除役廢棄物分類包裝審查技術，以利相關作業之執行。本篇報告是藉由美國核管會放射性物質運送包件標準審查計畫，闡述其用於指導管制單位執行放射性物質運送包件安全審查，確保管制單位審查品質的一致性；該標準審查計畫綜整相關法規資訊、審查程序及接受標準，以協助申照者發展放射性物質運送包件。本篇報告是藉由此標準審查計畫，藉由相關放射性物質運送包件審查經驗，作為未來國內審查相關放射性廢棄物盛裝容器之借鏡。

關鍵字：除役、低放射性廢棄物、包裝審查

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# 目　　錄

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# 1. 前言

## 1.1 研究目的

本報告為行政院原子能委員會放射性物料管理局委託核能研究所執行「103 年度精進放射性物料安全管制技術發展」計畫所屬子項計畫「核子反應器設施除役技術研究」之研究工作成果之一，期能提供除役產生之廢棄物分類包裝審查技術之參考。

## 1.2 研究內容

子項計畫「核子反應器設施除役技術研究」於 103 年規劃完成「除役廢棄物分類包裝審查技術研究」報告一冊。研究內容包含：

(1)國內外放射性廢棄物包裝容器法規研究：研究分析我國、國際核能機構(IAEA)與主要核能國家對於放射性廢棄物包裝審查之法規規定。

(2)國際放射性廢棄物分類包裝審查經驗研析：研究分析國際上主要核能國家放射性廢棄物分類包裝容器之選用及審查技術經驗。

(3)放射性廢棄物分類包裝審查經驗回饋：根據前述資訊與經驗分析結果，提出結論及建議。

## 1.3 報告架構

報告內容第 1 章說明研究目的、研究內容與報告架構；第 2 章說明國內外除役廢棄物相關法規；第 3 章為說明除役廢棄物標準審查計畫；第 4 章為為結論及建議。另外，包括參考文獻及附錄。

# 2. 除役廢棄物相關法規蒐集

核能電廠除役期間，除了用過核子燃料之外，將產生大量的低放射性廢棄物，這些低放射性廢棄物，一般可歸類為三種，包括：中子活化廢棄物、放射性污染廢棄物、以及其他放射性廢棄物。

中子活化廢棄物是指反應器爐心周圍受中子照射過的各類組件，包括反應器壓力槽、爐心內組件、結構鋼及反應器壓力槽周圍的混凝土。由於這些組件及材料被中子輻射照射過，整個結構內材料均被活化，無法利用表面除污方式來降低其活度。根據美國核管會 10 CFR 61.56 低放射性廢棄物分類方式，PWR 及 BWR 電廠若採取立即拆除之除役方式，其各種中子活化廢棄物之分類，大部份的中子活化廢棄物均可歸類為 A 或 B 類。對於 PWR 電廠而言，頂部爐心格板及底部支柱由於含有高濃度的 Ni-63 及 Nb-94 而歸類為 C 類。底部爐心筒，熱屏蔽，底部格板，及爐心側板所含的 Ni-59，Ni-63，及 Nb-94 之濃度超出 C 類之限值而被歸類為超 C 類。對於 BWR 電廠而言，其控制棒及爐內儀器、噴射泵、以及頂部燃料導架所含的 Ni-63 及 Nb-94 之活性在 C 類範圍內而被歸類為 C 類。爐心側板所含的 Ni-59、Ni-63、及 Nb-94 之活性則超過 C 類之最大限值，因此被歸類為超 C 類。由於這些核種之半衰期很長，即使封存 100 年後再進行拆廠，其所含的活性仍不可能顯著衰減，仍可能維持在超 C 類或 C 類之狀態。

核能電廠放射性污染廢棄物之污染源有兩種，第一種是腐蝕性產物，細微的金屬粒子從反應器冷卻、冷凝、及飼水系統的管路內壁剝落，然後流經爐心受中子照射後變成具有放射性，這些活化後的腐蝕產物將散布到反應器冷卻系統連接的所有系統。腐蝕性產物在系統管路中積存的數量，

視系統管路中液體的流速、溫度、及系統的幾何形狀等因素而定。如果這些系統有滲漏的話，則污染可能散布到地板、洩水坑、泥土等其他地方。第二種可能的污染源為燃料破損而造成少量的分裂產物外釋到反應器冷卻系統中，然後再流散到其他相連系統，並可能由於管路滲漏而污染地板、牆面、洩水坑及土壤等。

PWR 核能電廠除役，其污染性廢棄物幾乎包括圍阻體、燃料、廢棄物、控制廠房的所有組件、管件及設備，以及這些建築物的表面。依據美國核管會之估計，為了除污之目的，這些建築物混凝土表面必須敲掉約 50 mm 的深度。

污染性廢棄物所含的核種主要為半衰期較短之核種，例如 Co-60(半衰期 5.3 年)及 Cs-137(半衰期 30.0 年)，而長半衰期核種則極微量，因此一般均歸類為 A 類廢棄物。

核能電廠除役在進行拆除設備及廠房前，需先進行除污作業把廠內各系統內所積存的放射性廢液加以處理後排放掉。這些直接因除役動作所產生的除役廢棄物稱為放射性廢棄物，包含濕性廢棄物及乾性廢棄物，均歸類為其他放射性廢棄物。濕性廢棄物為處理除污溶液及電廠內各系統已有的放射性廢液所造成，包含濃縮廢液、過濾器殘渣、廢樹脂，以及各種除污過程所產生的中性化學溶液。這些濕性廢棄物將用水泥固化後裝桶。乾性廢棄物則包含丟棄的污染材料，如除污布、拖把、吸附紙、塑膠套、除污工具、及防範污染所用的衣物。

一般而言，在核能電廠除役期間，各項作業將產生為數不少的放射性廢棄物，依據放射性廢棄物之性質及來源，以及放射性廢棄物活度強弱之差異，採取之處理策略，例如：貯存、長期貯存、運送及處置等方式及目的，提供適合的放射性廢棄物盛裝或包封容器，將這些放射性廢棄物妥善

的包裝貯存，運送放置於安全無虞的場所，才能防止其污染環境或危害到人體的健康安全。國內核電廠在處理低放射性廢棄物的作法是先將放射性廢棄物轉變為較穩定的狀況，再以鍍鋅鋼桶容器盛裝，並遵循歐美等先進國家之處理方式加以嚴格管制，以避免低放射性廢棄物之洩漏或擴散。

美國電廠廢棄物容器的選用可依循下述幾項重要指標作出選擇，如(1)依照美國核管會(U.S. Nuclear Regulatory Commission，NRC) 10 CFR 法規的要求、(2)滿足最終處置場對於廢棄物容器的允收標準、(3)運送廢棄物容器時應符合美國交通部(Department of Transportation，DOT)法規要求、(4)在貯存期間應保持廢棄物容器的安全性與完整性等。

另外，NRC 與 DOT 均針對貯存用及處置用的廢棄物容器、廢棄物裝箱與運送廢棄物容器到處置場訂定出相關法規，並且所有廢棄物容器應滿足法規的管制與規定。NRC 在 10 CFR 61.56 提供了低放射性廢棄物類型和處置廢棄物封裝等規定：

(1) 廢棄物不可用紙板或纖維板包裝。

(2) 液態廢棄物應該先固化或使用吸水性材料來阻隔再封裝。

(3) 含有液態的固體廢棄物應盡可能減少腐蝕性液體，液體量不超過總體1%，。

(4) 廢棄物應確保不會有爆炸分解或在常溫常壓下與水生成反應的危險。

(5) 在運送、吊掛、處置廢棄物時，廢棄物不應該產生會危害人類的有毒氣體或煙霧。

(6) 廢燃料不能起火燃燒，起火材料應阻隔並且採用防火封裝。

(7) 氣態廢棄物應該在不超過1.5 大氣壓力下且20℃裝箱，廢棄物容器內的總活度不可超過100 居里。

(8) 有致病性、感染性的廢棄物應該作降低其最大潛在的危險處理。

在 10 CFR 61.56 附加的規定中，也加強限制 B、C 類廢棄物應該保持穩定性，避免在處置場產生衰竭或瓦解，這將會導致放射性核種的移動，然而置放期間最少要長達 300 年並且仍然保持其完整性。

EPRI “Guide for Operating an Interim On-Site Low-Level Radioactive Waste Storage Facility”技術報告中描述有關盛裝容器的設計與試驗指引中說明，低放射性廢棄物(LLW)貯存盛裝容器之關鍵設計需求、10 CFR 61.56 之規範、以及分類技術原則(Branch Technical Position, BTP)：

2.1 49 CFR 173.410

一般包件低放射性廢棄物貯存容器設計需求是依 49 CFR 173.410 之規範， 下列一般性規範適用於廢棄物貯存盛裝容器(Containers)：

(1) 盛裝容器可被簡單地吊舉搬運（handled）。

(2) 盛裝容器吊舉附屬物可以承載盛裝容器3倍的總重量(gross weight)。

(3) 吊舉附屬物必須經過試驗，驗證至貯存期限終止時仍可使用；亦即廢棄物盛裝容器於貯存貯限時可以安全地再取出（safely retrievable）。

(4) 盛裝容器必須避免過多突起，容易去污。

(5) 盛裝容器之設計應避免水分聚積在表面。

(6) 盛裝容器應避免不安全的附加物。

(7) 盛裝物質應與容器材質相容。

應注意有關內襯鋼桶(steel liners)仍未經過試驗以符合 49 CFR 173 及 10 CFR 71.71 中有關第二型工業包件(IP-2)之合格標準，但是仍可在密封鋼桶(Casks)中運送。已在處置場址內經固化的或 A 類的廢棄物，目前並無任何跡象顯示現行接受標準有任何問題。然而，相關設施應嚴肅考慮對盛裝容器增加防鏽塗層以延長其更長期的完整性。

美國核管會(USNRC) 的 10 CFR 61.56 規範，提供了最終處置用廢棄物盛裝的法規基礎，以及處置場廢棄物吊舉的最小需求，其廢棄物盛裝容器規範亦適用於貯存。另外，美國核管會分類技術原則(BTP)中更深入闡述其對於廢棄物包裝的立場，以下之章節定義了美國核管會所有對處置廢棄物容量的關鍵需求，也適用於現地貯存(on-site storage)。

2.2 10 CFR 61.56

最終處置之廢棄物不能使用紙板(cardboard)及纖維板(fiberboard)製作的容器盛裝，主要是針對乾性放射性廢棄物(dry active waste, DAW)。

2.3 USNRC 分類技術原則

- 盛裝容器應可抗由輻射效應(radiation effects)所產生的腐蝕劣化。
- 盛裝容器應可抗生物劣化(biodegradation)。
- 盛裝容器在最終處置環境下，承受壓力承載應可維持穩定狀況。
- 盛裝容器在最終處置後，暴露於潮濕及水環境下可維持穩定狀況。
- 廢棄物(as-generated waste)應可與盛裝容器相容。

以上法規也對提供測試以核發廢棄物形式證明，但是美國核管會已停止出版低放廢棄物(LLW)相關的專題報告(topical reports)，也不再核發廢棄物形式證明。廢棄物形式之申照審查現在必須由各州或美國能源部成立在愛達荷實驗室的輻射控制計劃協會(Conference of radiation control program directors, CRCPD)E-5 委員會所負責，依據分類技術原則的詳細高完整容器設計和接受標準辦理。

## 2.4 放射性物質安全運送規則

我國放射性物質安全運送規則第十六條規定，放射性物質各包件之設計、製造、試驗、文件建檔、使用、維護與檢查，以及運送與運送中之貯存等作業，均應建立品質保證計畫，以保證各項作業均能符合本規則之規定。

第十九條規定放射性物質依其型式，分為低比活度物質、特殊型式放射性物質、低擴散性放射性物質、含有可分裂物質及六氟化鈾等。包件以其盛裝放射性包容物之數量、性質及包裝之設計，分為甲型、乙型、丙型、工業、微量包件五種；包件含有可分裂物質或六氟化鈾者，應符合相關規定。含六氟化鈾之包件並應符合含有可分裂物質包件之管制相關規定。放射性物質、包裝及包件應符合附件三及附件四之相關規定。

## 2.5 放射性廢棄物處理貯存及其設施安全管理規則

我國放射性廢棄物處理貯存及其設施安全管理規則第八條規定，盛裝容器應符合下列規定：

一、材質、設計及製造，能防止腐蝕與劣化，並可確保設計年限內結構之完整。

二、考量操作及搬運之便利。

三、機械強度足以承受吊卸、搬運、貯存或最終處置等作業之負載。

四、容器封蓋及緊固設備，具操作之便利性，在吊卸及搬運過程中不致動搖或脫落。

五、容器外表應平整、易於除污並避免頂部積水。

第九條規定，盛裝容器經核准後始可使用，申請者應提出載明下列事項之報告，報請主管機關審核：

一、適用範圍。

二、設計基準、詳細工程設計及圖說。

三、容器材質、組成、尺寸、製造及防蝕方式。

四、試驗方法、標準及結果。

五、品質保證。

六、其他經主管機關指定之事項。

第十二條規定裝有放射性廢棄物之盛裝容器表面，應有輻射示警標誌及編號。輻射示警標誌之中心圓半徑不得小於二公分。

## 2.6 低放射性廢棄物最終處置及其設施安全管理規則

我國「低放射性廢棄物最終處置及其設施安全管理規則」第三條規定低放射性廢棄物依其放射性核種濃度分類規定如下：

一、A類廢棄物：指低放射性廢棄物所含核種濃度低於（含）表2.6.1濃度值之十分之一倍及低於（含）附表2.6.2第一行之濃度值者；或

廢棄物所含核種均未列入附表一及附表二者。

二、B類廢棄物：指低放射性廢棄物所含核種濃度高於附表二第一行之濃度值且低於（含）第二行之濃度值者。

三、C類廢棄物：指低放射性廢棄物所含核種濃度高於附表一濃度值十分之一倍且低於（含）表2.6.1之濃度值者；或高於表2.6.2第二行之濃度值且低於（含）第三行之濃度值者。

四、超C類廢棄物：指低放射性廢棄物所含核種濃度高於表2.6.1之濃度值者；或高於表2.6.2第三行之濃度值者。

第四條規定低放射性廢棄物最終處置，應依下列規定：

一、A類廢棄物應符合第五條之規定。A類廢棄物與B類廢棄物或C類廢棄物混合處置者，應符合B類廢棄物或C類廢棄物之相關規定。

二、B類廢棄物應固化包裝，其廢棄物應符合第五條及第六條之規定。B類廢棄物與C類廢棄物混合處置者，應符合C類廢棄物之相關規定。

三、C類廢棄物應固化包裝，其廢棄物除符合第五條及第六條之規定外，應加強處置區之工程設計，以保障監管後誤入者之安全。

四、超C類廢棄物非經主管機關核准，不得於低放處置設施進行處置。

未固化之A 類廢棄物，應盛裝於經主管機關核准至少能維持一百年結構完整之容器或封存於具相同容器功能之工程結構中進行處置。不適合固

化或經固化未達品質要求之B類廢棄物及C類廢棄物，得以經主管機關核准之高完整性容器盛裝進行處置。

表2.6.1 單一長半化期核種濃度值

| 核種 | 濃度值 |
|---|---|
| $^{14}C$ | 0.30 $TBq/m^3$ |
| $^{14}C$（活化金屬內） | 3.0 $TBq/m^3$ |
| $^{59}Ni$（活化金屬內） | 8.1 $TBq/m^3$ |
| $^{94}Nb$（活化金屬內） | 0.0074 $TBq/m^3$ |
| $^{99}Tc$ | 0.11 $TBq/m^3$ |
| $^{129}I$ | 0.0030 $TBq/m^3$ |
| TRU(半化期大於5 年之超鈾阿伐放射核種) | 3.7 kBq/g |
| $^{241}Pu$ | 130 kBq/g |
| $^{242}Cm$ | 740 kBq/g |

表 2.6.2 單一短半化期核種濃度值

| 核　　種 | 濃度值（$TBq/m^3$） | | |
|---|---|---|---|
| | 第一行 | 第二行 | 第三行 |
| 半化期小於 5 年之所有核種總和 | 26 | 註一 | 註一 |
| $^{3}H$ | 1.5 | 註一 | 註一 |
| $^{60}Co$ | 26 | 註一 | 註一 |
| $^{63}Ni$ | 0.13 | 2.6 | 26 |
| $^{63}Ni$（活化金屬內） | 1.3 | 26 | 260 |
| $^{90}Sr$ | 0.0015 | 5.6 | 260 |
| $^{137}Cs$ | 0.037 | 1.6 | 170 |

註一：B類廢棄物及C類廢棄物並無此核種濃度值之限制。可從實際執行運送、吊卸與最終處置作業時，考量體外輻射與衰變熱，而限制這些核種之濃度。除非由本表內其他核種決定廢棄物歸於C類廢棄物，否則應歸於B類廢棄物。

註二：多核種之分類：

若低放射性廢棄物中含有多核種時，其分類應按下式判斷。

$$\sum_{i=1}^{n} \frac{C_i}{C_{i,0}} \leq 1$$

式中

$C_i$：第 i 個核種之濃度。

$C_{i,0}$：第 i 個核種第 0（0=A,B,C）類之濃度值。

n：所含核種之數目。

若滿足上式，則可歸為第 0（0=A,B,C）類廢棄物。

由於除役放射性廢棄物盛裝容器及包件因不同目的及效能，具有極大的差異性，若規劃為運輸用之盛裝容器，則需依據「放射性物質安全運送規則」之規定辦理，若要申請處置用容器時，則可依據「低放射性廢棄物盛裝容器使用申請書導則」及「低放射性廢棄物最終處置盛裝容器審查規範」等相關法規辦理，申照時提出盛裝容器的相關資訊，下一章將參考美國核管會有關放射性廢棄物標準審查計畫之內容，以瞭解美國對相關放射性廢器物盛裝容器之審查過程及相關作法。

## 3. 運送包件標準審查計畫

為了建立除役放射性廢棄物分類包裝審查技術，藉由美國核管會針對放射性物質運送包件所建立之標準審查計畫(Standard Review Plan for Transportation Packages for Radioactive Material)以瞭解美國在執行放射性廢棄物包件申照審查相關業務時之做法。美國核管會放射性物質運送包件標準審查計畫具有三個主要目的，包括：

- 綜整盛裝容器核准(package approval)所需法規要求。
- 說明核管會相關法規要求程序。
- 將核管會發展之盛裝容器認證(package certifications)業務(practices)文件化。

標準審查計畫可輔助美國核管會法規指引(Regulatory Guide, RG) 7.9 第 1 版及第 2 版之規定，提供申照者(applicants)標準的放射性物質包件申請格式和內容，申請包件及盛裝容器認證執照。由於放射性廢棄物盛裝容器及包件因不同目的及效能，具有極大的差異性，以及有許多不同的方法可以評估盛裝容器的設計，所以沒有一個審查計畫可以詳細的包括每一種申請狀況，因此需要修改及擴充審查計畫的相關指引來適應特定的盛裝容器設計。

此審查計畫在每一個章節中的格式和 RG 7.9 建議的申請書類似，並著重在審查技術和法規基礎上、審查完成的方式、以及符合核准標準的一般

性適用的盛裝容器的審查結果。美國核管會針對放射性物料運送盛裝容器建立的的標準審查計畫，包括下列章節：

- 一般資訊審查(GENERAL INFORMATION REVIEW)
- 結構審查(STRUCTURAL REVIEW)
- 熱傳審查(THERMAL REVIEW)
- 包封容器審查(CONTAINMENT REVIEW)
- 屏蔽審查(SHIELDING REVIEW)
- 臨界審查(CRITICALITY REVIEW)
- 操作程序審查(OPERATING PROCEDURES REVIEW)
- 驗收試驗和維護方案審查(ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW)

一般而言，標準審查計畫的審查領域會與 RG 7.9 的主要章節相對應，在法規要求章節中，概述 10 CFR 71 適用之法規需求，在許多事例中，法規中的文字敘述都較簡短，有時會合併二項或以上的相關規定以求簡潔。然而，在文中措詞的修改並無意要改變或解釋的法規規定。藉由參考並驗證其他標準來證明盛裝容器符合法規要求，並說明相關接受標準。審查程序(Review Procedures)會提供盛裝容器審查程序的指引，和審查領域互相平行。因為許多不同的盛裝容器設計，審查人員可能需要擴大或修改這些程序，以適應特定的盛裝容器，或引用申請書中的評估方法。盛裝容器申請書的核准，沒有任何一部份是和其他部份的資訊獨立審查完成的。例如，臨界評估部份是取決於(1)盛裝容器和內容物的一般性資訊描述，和(2)假設性意外情形測試下結構和熱傳評估的盛裝容器狀況。同樣地，臨界評估的

結果可能會導致需要採取特別的操作程序或驗收測試，因此，審查計劃的每一個審查程序，提出了界面的示意圖，這些示意圖只是一個例子，特定的界面可能會因特殊的盛裝容器設計而改變。審查結果都記錄在安全評估報告(Safety Evaluation Report)，該報告總結了以下三點：

- 適用的法規需求
- 申請書中的方法顯示滿足法規需求
- Staff's review of the evaluation presented in the application.

其評估結果(Evaluation Findings)提出可在安全分析報告中說明主要評估結果的例子，審查人員將修改措辭，以恰如其份的引導申請書和審查方法的具體細節。

標準審查計畫的附錄中，提供了 8 個美國核管會審查的不同型式放射性物料盛裝容器的詳細資訊，藉由確認每一種盛裝容器典型的關鍵安全功能(key safety features)和主要審查領域(principal areas of review)的案例，來補充審查計畫的資訊。8 個審查案例包括：

- 放射性照相廢棄物盛裝容器(RADIOGRAPHY PACKAGES)
- B型廢棄物盛裝容器(TYPE B WASTE PACKAGES)
- 未照射燃料盛裝容器(UNIRRADIATED FUEL PACKAGES)
- 低濃縮氧化鈾盛裝容器(LOW ENRICHED URANIUM OXIDE PACKAGES)
- 超鈾廢棄物盛裝容器(TRANSURANIC WASTE PACKAGES)
- 低濃縮六氟化鈾盛裝容器(LOW ENRICHED URANIUM HEXAFLUORIDE PACKAGES)
- 高濃縮鈾或鈽盛裝容器(HIGH ENRICHED URANIUM OR

PLUTONIUM PACKAGES)

- B型特殊型式盛裝容器(TYPE B SPECIAL FORM PACKAGES)

## 3.1 一般資訊審查(GENERAL INFORMATION REVIEW)

### 3.1.1審查目的

一般資訊審查目的是要確認包件設計已詳細描述，可提供適當的基礎來做評估審查。

### 3.1.2審查範圍

審查範圍(Areas of Review)必須包括包件設計的描述和工程設計圖，並包括下列事項：

- 簡介(Introduction)：包括應用目的及摘要資訊。
- 盛裝容器描述(Package Description)：包括盛裝方式(Packaging)、包封容器邊界(Containment Boundary)、盛裝內容物(Contents)、操作功能(Operational Features)。
- 盛裝容器之一般需求(General Requirements for All Packages)：包括最小尺寸(Minimum Size)、開啟指示功能(Tamper-Indicating Feature)。
- 設計圖(Drawings)。

### 3.1.3法規要求(Regulatory Requirements)

10 CFR 71適用於一般資訊的法規要求，包括：

- 申請書必須詳細地及充分的描述包件之設計，以提供適當評估基礎。[§71.31(a)(1), §71.33(a)]

- 申請書必須詳細及充分的描述內容物，以提供適當的包件器評估基礎。[§71.31(a)(1), §71.33(b)]
- 申請書必須參考申照者經核管會核准之品質保證計畫。[§71.31(a)(3), §71.37]
- 申請書必須確定可適用於包件的設計、製造、組裝、測試、維護和使用的規範和標準。[§71.31(c)]
- 已核准包件之執照更新必須於到期日30天前遞交申請書，以確保盛裝容器可繼續使用。[§71.38]
- 所有包件核准條件的變更，都必須經由核管會核准。對已核准包件的修改申請，可能會受到§71.13和§71.31(b)的規定管制。[§71.107(c)]
- 包件的最小外形尺寸不得小於10厘米（4英寸）。[§71.43(a)]
- 包件的外部，當完整無損時，必須具備經認證，未授權人員無法打開之功能。[§71.43(b)]
- 包件裝有超過0.74 TBq(20 Ci)的鈽時，必須滿足有關鈽的特殊污染要求。[§71.63]
- 可分裂物質之包件(a fissile material package)必須指定運輸參數(transport index)做為臨界控制，以限制單一運送中盛裝容器的數量。[§71.59, §71.35(b)]
- 運輸指數大於10盛裝容器，必須專用運送(exclusive-use shipment)。[§71.47(b), §71.59(c)]

3.1.4接受標準(Acceptance Criteria)

- 包件(package)必須符合第3.1.3節中綜合之法規要求。

- 包件設計和操作必須有詳盡的細節以提供足夠的說明，以便依據10 CFR 71進行評估審查。設計必須以符合規定之工程設計圖顯示。

### 3.1.5審查程序(Review Procedures)

審查應確保包件設計和操作的基本資訊說明充分且詳細，使得包件效能得以在申請書的各章節中評估。如圖3-1所示，在一般資訊章節中顯示與其他章節的資訊架構，以及後續的章節中，一般資訊與包件相關各項描述和評估審查的關係。

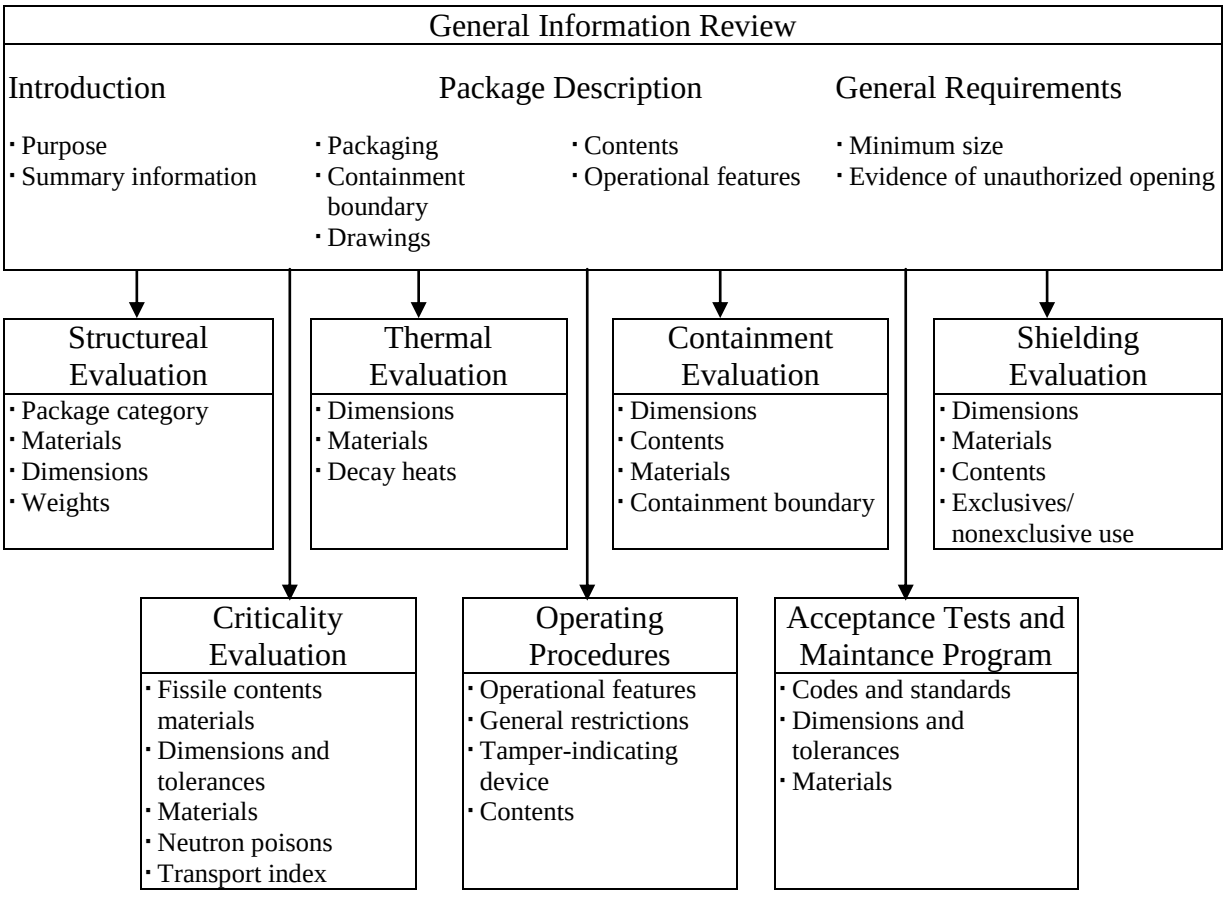


圖3-1 一般資訊與其他章節審查之資訊架構

3.1.5.1 簡介(Introduction)

3.1.5.1.1 申請目的(Purpose of Application)

申請目的應明確說明，可能是申請一項新的設計、修改已核准之設計、或是執照更新。申請核准新的設計應該要提供完整，及包含在10 CFR 71 Subpart D中定義的資訊。

申請已核准的設計修改，應明確定義所請求的變更，包括設計的變更、授權內容的變更、或核准條件的變更，申請變更的內容應該也要明確的在包件設計圖中標示出來。不符合NCR核准認可圖樣(drawings)的包裝(packagings)不會取得§71.12一般許可之授權使用，同樣地，只有在批准的特定內容可以傳送。包件操作程序(package operating procedures)、接受試驗、及維修方案(maintenance program)也可以被指定做為核准的條件。

申請已核准的設計修改，應包括請求變更和依據的評估，這些變更不會影響到包件的性能以滿足10 CFR第71法規的要求。如果適用的話，申請修改的部份也受到§71.13及§71.38的規範。

現有執照更新申請，應該在到期30天前執行，以確保可繼續使用。

3.1.5.1.2 概要資訊(Summary Information)

驗證申請書引用申照人經NRC批准之品質保證計劃。確認包件的類型和型號被指定。一個新的乙型包件(Type B

package)設計將被指定為B(U)-85，除非它有超過700千帕(100 psi)表壓的最大正常工作壓力，或是洩壓裝置依據§71.38(假設意外條件)規定的測試下允許放射性物質釋放，如此，包件將被指定為B(M)-85。

審查包件的預期用途和內容物的最大活性(activity)，確保它們與指定的包件類型(package type)一致。對於乙型包件，驗證指定的包件類型是正確合理的，包件類型的定義總結於表1-1，詳細的判斷包括計算包裝內容物最大活度的有效A1及A2值，可能在附錄中或其他章節中展示。基於包件類型分級，確認適當的ASME法規(ASME 1995)或其他標準(NUREG/CR- 3019; NUREG/CR-3854)，被指定用來評估會影響包封容器(containmnet)結構完整性、臨界、或屏蔽的組件。

表1.1 乙型包件類型分級(RG 7.11)

| Contents Form/ Category | Category I | Category II | Category III |
|---|---|---|---|
| Special Form | Greater than 3,000 A1 or greater than 1.11 PBq (30,000 Ci) | Between 3,000 A1 and 30 A1, and not greater than 1.11 PBq (30,000 Ci) | Less than 30 A1 and less than 1.11 PBq (30,000 Ci) |
| Normal Form | Greater than 3,000 A2 or greater than 1.11 PBq (30,000 Ci) | Between 3,000 A2 and 30 A2, and not greater than 1.11 PBq (30,000 Ci) | Less than 30 A2 and less than 1.11 PBq (30,000 Ci) |

對於可分裂物質包件，確認每一項包裝內容物之臨界運送指數。基於運送指數，決定包件是否需要限制專用交運(exclusive-use shipment)。

3.1.5.2 包件說明(Package Description)

3.1.5.2.1 包裝(Packaging)

審查包件的文字描述，以及其他應適當提供的概述、草圖、或其他概圖。包件的工程設計圖應要在附錄中提供，並確認下列資訊應要充分的說明討論：

- 一般包裝說明，包括整體尺寸、最大重量、以及最小重量。
- 包封特性(Containment features)。
- 中子和γ屏蔽功能，包括人員屏蔽(personel barriers)。
- 臨界控制功能，包括中子毒物、緩和劑(moderators)、格架(spacers)等。
- 結構功能，例如吊升及固定設施、防撞緩衝器或其他防護設施、內部支撐或定位功能、外殼或外包裝、包裝封閉設備等。
- 熱傳特性(Heat transfer features)。
- 包裝標誌(Packaging markings)。

依據10 CFR 2.790的規定，一些包件的專有資訊(proprietary information)，例如：工程設計圖中顯示的特殊設計細節，將被限制不能公開揭露。限制不能公開揭露

(withholding)的要求必須有保密切結(affidavit)，而且必須包括必要的資訊，來支持材料是專有的權利(proprietary)要求。不公開揭露的要求是由總法律顧問辦公室審查，以符合10 CFR2.790的規定。

3.1.5.2.2 包封邊界(Containment Boundary)

審查的目的在確認申請書所定義之包封系統的確切邊界，包括密封槽(containment vessel)、銲接(welds)、注排水口(drain or fill ports)、閥門(valves)、密封(seals)、測試端口(test ports)、洩壓裝置(pressure relief devices)、端塞(lids)、封蓋(cover)、蓋板(plates)，和其他密封裝置等。假如單一密封包件使用多重密封裝置，則此密封裝置定義為包封系統密封(containment-system seal)，應清楚的鑑定。並應提供包封系統之概圖(sketch)，以及在附錄中附上所有組件的工程設計圖。

如果包裝內容物包含超過0.74 TBq(20 Ci)的鈽，則包裝(packaging)必須具有內部及外部包封容器(containment system)，除非依§71.63之規定豁免。

3.1.5.2.3 內容物(Contents)

審查的目的在確認，對於包容物描述的細節與合格證明書(Certificate of compliance)相同。描述的細節至少應該包括下列各項資訊：

- 放射性物質（活度或質量）的識別和最大數量
- 可分裂物質的識別和最大數量

- 物理化學型態，包括密度和濕度，及其他緩和成分(moderating constituents)的存在。
- 包件內容物的位置(location)及配置(configuration)，包括二次容器(secondary containers)、包裝材料(wrapping)、支撐(shoring)、和其他沒有定義為包件一部分的材料。
- 使用做為中子吸收劑或緩和劑的非可裂物質的識別和數量。
- 任何易發生化學、電化學、或其他反應，包括會產生可燃性氣體的材料。
- 最大正常操作壓力。
- 適當的最大和最小重量。
- 最大衰變熱。

3.1.5.2.4 操作功能(Operational Features)

確認合適的操作功能已充份討論，如果可以的話，並應附上任何操作功能的流程圖。

3.1.5.3 所有包件的一般要求(General Requirements for All Packages)

確認包括符合下列法規要求§71.43 (General Requirements for All Packages)：

- 包件最小的整體尺寸不小於10公分(4英吋)。
- 包件的外部，當完整無損時，必須具備經認證，未授權人員無法打開之功能。[§71.43(b)]

3.1.5.4 附錄(Appendix)

驗證工程圖的資訊信息足夠詳盡，並與包件描述一致。依據§71.3之規定，放射性物質的運輸必須具有授權的執照。

在§71.12一般許可規定中，授權利用經NCR許可之包件運輸放射性物質，並要求領有執照之人(licensees)遵守一般許可證的規定，包括核准包件的期限和條件。包件若不符合NRC批准的設計圖，將不能取得授權使用。

確認設計圖中具有標題框，可驗證製造單位、圖號、表號、標題、日期、簽名或縮寫字，顯示設計圖已核准。修定後的設計圖應確認每一個更新版本之號碼、日期和說明。專有資訊(proprietary information)亦應清楚標識。設計圖應包括下列項目：

- 包件及內容物的一般性配置(general arrangement)，包括尺寸。
- 影響包件評估的設計功能。
- 包括標識。
- 包件最大允許重量。
- 內容物和二次包裝(secondary packaging)最大允許重量。
- 最小重量。

在適當的情況下，設計特徵的相關資訊應包括下列項目：

- 設計特徵及其組成之識別
- 結構材料，及其規格
- 法規，標準，或其進行加工、裝配和測試等類似的規範文件
- 相對其他包件特徵之立置

- 外形尺寸及適當公差
- 操作規範（例如：螺栓扭矩）
- 焊接設計和檢驗方法

NUREG/ CR-5502提供申請書提交有關工程圖的額外指引。確認附錄中包括適用的參考文獻及清單，一般是不會提供給評審。附錄中還應當提供特殊製程的資訊，包件種類的確認，以及其他適當的補充資訊。

3.1.6評估結果(Evaluation findings)

安全評估報告在申請書的一般資訊章節中，通常不包括具體結果(specific findings)。

## 3.2 結構審查(STRUCTURAL REVIEW)

### 3.2.1審查目的(Review Objective)

審查的目的是要驗證在正常運送狀況和假設意外事故狀況規定的試驗，包件設計的結構性能已被充分地評估。而且該包件具有足夠的結構完整性，以滿足10 CFR 71的法規要求。

### 3.2.2審查範圍(Areas of Review)

包件的結構設計審查應包括以下幾點：

#### 3.2.2.1 結構設計說明

- 一般資訊說明，包括重量和重心
- 規範和標準的鑑定

#### 3.2.2.2 材料

- 材料特性和規格
- 化學，電化學，或其它反應的防止
- 材料的輻射影響

3.2.2.3 製造與檢驗

- 製造
- 檢驗

3.2.2.4 包件吊舉與固定標準

3.2.2.5 綜合考量

- 試驗評估
- 分析評估
- 壓力

3.2.2.6 正常運送狀況

- 熱(Heat)
- 冷(Cold)
- 減少外部壓力
- 增加外部壓力
- 振動(Vibration)
- 噴水(Water Spray)
- 自由墜落(Free Drop)
- 角墜落(Corner Drop)
- 壓縮(Compression)
- 滲透(Penetration)

3.2.2.7 假設意外事故狀況

- 自由墜落
- 擠壓(Crush)
- 穿刺(Puncture)
- 熱(Thermal)
- 浸没－可分裂材料
- 浸没－所有包件

3.2.2.8 附錄

3.2.3法規要求(Regulatory Requirements)

10 CFR 71適用於結構審查的法規要求包括下列事項：

- 包件必須充分描述和評估，以證明其符合10 CFR 71對結構的法規要求。[§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- 包件的效能必須依據10 CFR 71.71的法規要求，在正常運輸條件下規定的試驗來評估。[§71.41(a)]
- 包件的效能必須依據10 CFR 71.73的法規要求，在假設意外事故狀況下規定的試驗來評估。[§71.41(a)]
- 申請書必須確認用於包件結構設計所依據的規範和標準。[§71.31(c)]
- 包件必須由保證不會有顯著的化學、電化學或其他反應的材料和結構製造，包括可能由於漏水進入包裝組件中、包件內容物中、或者包裝內容物及包件中所造成的反應。輻射對結構材料的影響亦必須考慮在內。[§71.43(d)]
- 依據§71.71的法規要求，在正常運送狀況的試驗下，包件經過設計、施工、及準備交運，必須不能有內容物的損耗或分散，亦

不會使得包裝的有效性顯著減少。[§71.43(f), §71.51(a)(1)]

- 包件設計必須符合§71.45的起吊和固定的法規要求。
- 包件設計必須具有足夠的結構完整性，以滿足§71.85(b)法規要求的內部壓力測試。

3.2.4接受標準(Acceptance Criteria)

- 包件必須符合第2.3節中列出的法規要求。
- 包件必須具有足夠的結構完整性，以滿足10 CFR 71規範的包封容器(containment)、屏蔽、次臨界和溫度的法規要求

3.2.5審查程序(Review Procedures)

結構審查應確保包件設計已充分描述和評估，證明包件具有足夠的結構完整性，以滿足10 CFR 71在正常運送狀況和假設意外事故狀況下的法規要求。結構審查是基於申請書中一般資訊及熱傳評估相關的說明和評估內容而進行審查，同樣地，在其他審查中，結構審查的結果亦會被考量,圖3-2顯示結構審查與各章節相對的資訊架構。結構審查確認描述結構設計功能之文字和概圖，與工程圖設計圖和結構的評估中使用的模型是一致的。

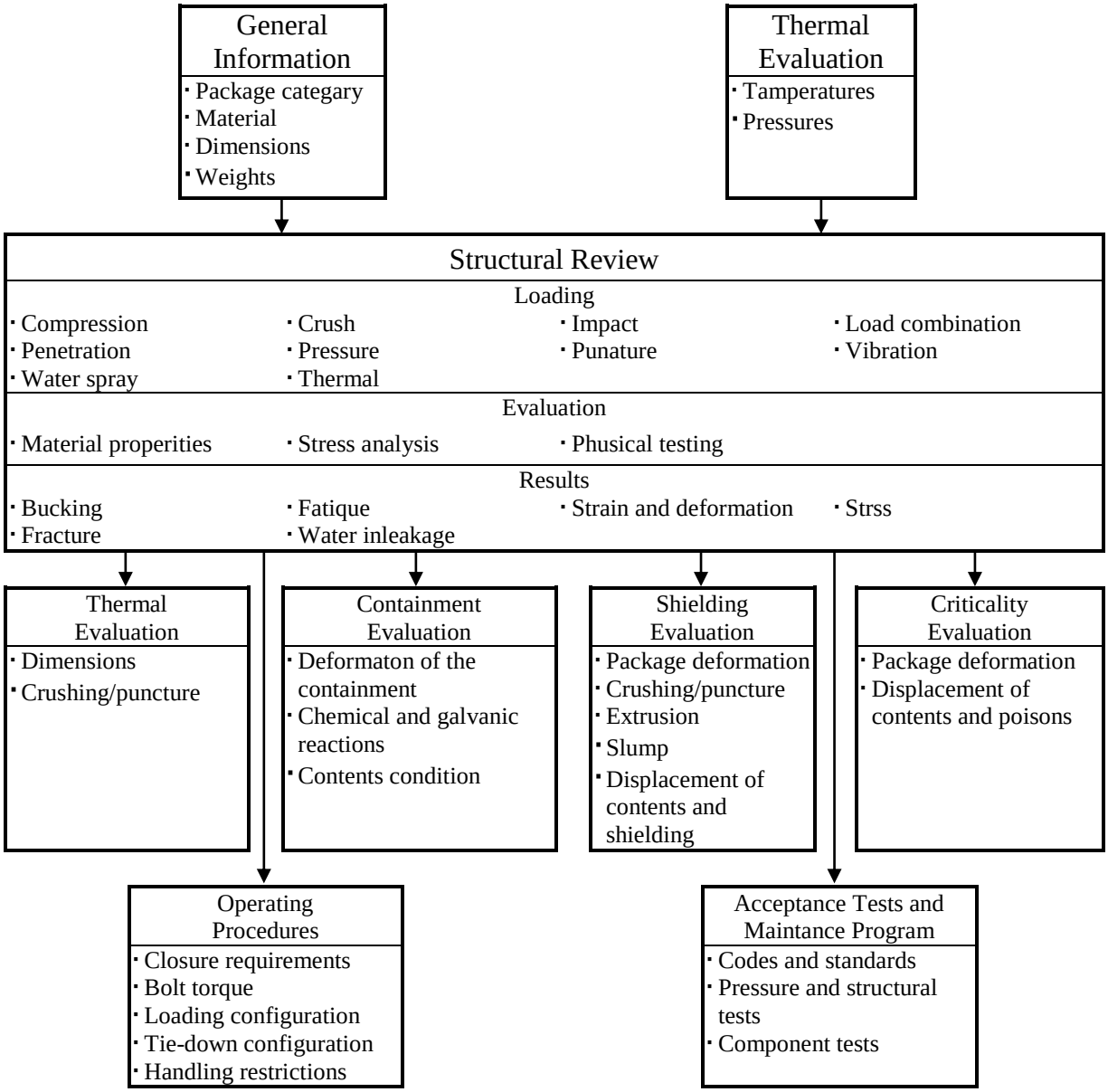


圖3-2 結構審查與各章節審查對應之資訊架構

### 3.2.5.1 包件設計規範和標準鑑定

審查申請人已確定的規範和標準。審查應包括申請人已確定的規範和標準的適用性評估。該評估可能包括以下各方面的考量：

- 規範或標準，如果沒有特別為了交運包件，是為了結構或類

似的設計和材料的特性審查而發展。

- 規範或標準是為了類似負載狀況下的結構審查而發展。
- 規範或標準是為有類似失敗的後果的結構審查而發展。
- 規範或標準充分解決潛在故障模式。
- 規範或標準的安全範圍。

該規範或標準應考慮包件類別（見表1.1）。NUREG/ CR-3854提供的工業規範和標準列表可被用來根據包件類別製造運送包裝組件。

法規指引(Regulatory guides)和NUREGs法規利用現有適用的規範和實務做法提供運送包件評估額外的設計指引。例如RG 7.8在綜整包件負載的設計評估。RG 7.6提供了乙型包件(Type B packages)包封容器系統(containment system)設計標準。NUREG/ CR-4554, Vol. 6提供儲存容器的翹曲(buckling)評估。NUREG/ CR-3019提出有關運輸包件的焊接標準。

3.2.5.2 材料

3.2.5.2.1 材料特性及規格

審查結構材料的性質，確認適用的規範及法規以掌握每種材料的性質。材料及其性質應該與所選擇的設計規範或標準是一致的。如果沒有可用的標準，申請書應該提供足夠的材料性質和規格的文件，以提供包件的設計和製造。

驗證結構組件材料性質具有足夠的斷裂韌性強度，在正常運輸狀況和假設意外事故狀況下，足以避免脆性斷裂發

生。RG 7.11和7.12提供了斷裂韌性強度標準。驗證材料特性適合負載的條件（例如：靜態或動態的衝擊負載，熱或冷的溫度，潮濕或乾燥狀況）。驗證適當的溫度下，允許的應力限制條件已被定義，且與最低和最高使用溫度是一致的。驗證防撞緩衝器受力變形特性的影響，是基於適當的測試條件和溫度狀況。

3.2.5.2.2 避免化學、電化學及其他反應

審查包件的材料和塗層，以驗證它們不會在包裝組件、內容物、以及包裝組件與內容物間，產生顯著化學或電化學反應。審查應考慮水的漏洩造成的反應。評估可能產生氫氣等易燃易爆氣體。考量金屬組件間可能因彼此的物理接觸，而產生電流的相互作用和共晶的形成。這種相互作用可能是乏鈾、鉛、或者鋁與鋼的接觸發生。

3.2.5.2.3 材料的輻射效應

輻射效應對包裝材料的任何破壞影響之驗證皆已適當考慮。這些影響包括密封、密封材料、塗層、粘合劑、和結構材料的劣化。

3.2.5.3 製造與檢驗

3.2.5.3.1 製造

一般製作項目：

- 成型、裝配、校準
- 焊接和銅焊
- 熱處理

- 機械節點。

如果製造規格已由可接受的規範或標準所規定（例如：ASME、AWS），則該規範或標準應在工程設計圖上標示（NUREG/ CR-5502）。除非申請書說明，否則用於設計的相同規範或標準，應該也可用於製造。對於沒有規範或標準適用的組件，申請書應確認其規範，其所得到的評估結果和描述的其控制方法，以確保這些規範可被達成。說明中可能參考了品質保證或其它適當的規範文件，皆應在工程設計圖上規定。正如本審查計劃第1節指出，工程設計圖一般包括核准的合格證明書項目。

3.2.5.3.2 檢驗

檢驗設定方法及標準，以確定製造成果是否可被接受。除非申請書特別註明，否則用於製造的相同規範或標準，應該也可用於檢驗。

對於沒有製造規範或標準適用的組件，申請書應該總結檢驗方法和驗收準則，在第3.8節驗收測試和維護程序中說明。正如本審查計劃第3.8節指出，驗收測試通常包括合格證明書的核准條件。

3.2.5.4 包件的吊舉及固定標準

3.2.5.4.1 吊舉設備(Lifting Devices)

審查包件結構組件有關吊舉設備之評估與設計，包括與包件本體的連接，以及包件本體與吊舉設備相連周圍的範圍，驗證及評估吊舉設備符合§71.45(a)的規定，包括在

過度負荷下所產生的失效狀況。

3.2.5.4.2 固定裝置(Tie-Down Devices)

審查屬於包件結構組件的固定裝置之評估與設計，包括其與包件本體的連接，以及包件本體與固定裝置周圍的範圍。驗證其評估固定裝置符合§71.45(b)的規定，包括在過度負荷下所產生的失效狀況。

3.2.5.5 一般注意事項(General Considerations)

相關的評估應證明包件的結構效能應符合第3.2.5.6節的正常運輸狀況和第3.2.5.7節的假設意外事故中討論的標準。

- 最極端的初始條件已經使用（參見RG7.8初始條件的選擇指引）。
- 評估方法適用於考量的負載荷條件，並依照公認的慣例和規則辦理。
- 評估結果的解釋是正確的。
- 最具破壞性的方向已經被考慮。一個組件最具破壞性的方向，可能不是另一個組件最具破壞性的方向。
- 設計標準，例如RG7.6所提供的規範已被應用。

3.2.5.5.1 試驗評估(Evaluation by Test)

如果包件是通過測試評估，審查應至少包括以下內容：

- 檢查用於自由下落和粉碎試驗表面的描述（例如：材料、質量、整體尺寸等）。確認它符合§71.73(c)(1)規定的基本堅硬的表面。

- 檢查用於穿刺測試鋼筋的描述（例如：材料、尺寸、方位、安裝方法）確認其牢固地連接至基本堅硬表面，具有足夠的長度以使在包件上產生最大傷害，並滿足§71.73(c)(3)其他規定。
- 驗證的試樣必須使用在設計中所指定相同的材料、方法和品質保證製造。任何差異都必須標識說明，並在該申請書進行影響評估。包件內容物的替代品應該與實際的內容物具有相同的代表重量。
- 確認所選擇的墜落方位，是考量其為預期最大傷害的方位，並且確認該選擇是合理的。
- 確認所有的試驗結果都已進行評估，以及他們的含義皆可被闡釋，包括測試物件內部和外部的損害。意外的或無法解釋的試驗結果，顯示測試可能發生問題，或不可再現的試樣的行為，都應該提出討論和重新評估。
- 如果有的話，檢查試驗的影像和照片。
- 驗證試驗顯示其安全性有足夠餘裕。測試結果應清楚地表明，該試驗的結果可以可靠地再現。驗證試驗結果的描述中包括的機械特性，試驗條件和診斷程序的不確定性的影響的討論。
- 審查評估通過/失敗測試條件的標準。比較試驗結果與標準。

3.2.5.5.2 分析評估(Evaluation by Analysis)

如果申請書提供分析評估，審查應至少包括以下內容：

- 驗證分析計算，以及所有的假設都有清楚的說明（見RG 7.6分析之設計標準指引）。
- 驗證包件所顯示的負載反應，就應力和應變對組件和結構元件而言，視情況評估個別元件的結構穩定。
- 驗證分析方法考量在任何角度、剛體轉動、和二次碰撞的影響。
- 如果使用電腦程式碼，驗證目標應用是有效的，以及使用的方法與標準方法和程序一致，並且經過基準化。
- 如果準靜態分析技術已被使用，驗證動態放大因子已經適當應用。NUREG/ CR-3966提供衝擊影響分析準靜態和動態分析方法之說明。
- 驗證模式和材料特性適合考量的負載組合。確保該材料性質（例如：彈性、塑性）與分析方法一致。申請書應該證明在該材料性質的應變率。確認該分析已考量實際應力-應變或工程應力-應變的結果。
- 審查分析結果的匯總表，將結果與規定的驗收標準進行比較，並確認符合驗收標準。驗證所有的驗收標準都與相應的規範和標準一致。

每個具有超過35 kPa(5 psi)最大正常操作壓力的包件，第一次使用之前，容器密封系統必須依據§71.85(b)之規定，進行其最大正常操作壓力150％的壓力試驗，以確認申請書中提供的驗收測試分析。

3.2.5.6　正常運送狀況下之結構評估

在正常運輸狀況下，依據10 CFR 71.71規定的試驗和條件進行包件的評估，這些試驗必須不會導致包件的有效性的降低。例如，應該有：

- 無包件內容物任何的損耗或擴散。
- 無包件結構變化致減少組件對屏蔽、熱傳、維持次臨界、或密封所需之有效性。
- 無變更致影響包件承受假設意外事故狀況的能力。

試驗前後的氣溫必須保持在-29℃(-20℉)到+38℃(100℉)之間，這對於所考慮的特徵是最不利的值。在儲存容器內的初始內壓必須是最大正常操作壓力，除非較低的內部壓力與所選擇的環境溫度是不太有利的。

3.2.5.6.1　熱(Heat)

確認在熱狀態下，熱傳效能和最高溫度的評估，與熱評估章節一致。該評估應考慮結合最大內部熱負載，和任何殘餘製造應力的最大正常工作壓力。驗證任何細微的熱擴展和可能的幾何干擾都被考慮在內。如果結構設計經過工程分析評估，驗證在正常負載狀況下，應力在限制範圍內。

3.2.5.6.2　冷(Cold)

確認在冷測試條件下，熱傳效能和溫度的評估，與熱傳章節一致。

評估應考慮結合最小內部熱負載（通常假定為沒有衰

變熱）以及任何殘餘製造應力的最小內部壓力。驗證任何細微可能會導致幾何干擾的熱擴展，都被考慮在內。驗證可能的液體凍結都已被考慮。驗證在正常負載狀況下，應力在限制範圍內。

3.2.5.6.3 減少外部壓力(Reduced External Pressure)

確定申請書已充分評估減少外部絕對壓力至25 kPa(3.5psi)，對包件設計的影響。驗證申請書已考量包件內部和外部間，以及密封系統內部和外部之間可能的最大壓力差。

3.2.5.6.4 增加外部壓力(Increased External Pressure)

確定申請書已充分評估增加外部壓力至140 kPa(20 psi)，對包件設計的影響。驗證申請已考量結合最小的內部壓力之負載條件。驗證申請書已考量包件內部和外部間，以及密封系統內部和外部之間，最大可能的壓力差。考量翹曲的可能性(見NUREG/ CR-4554，Vol. 6)。

3.2.5.6.5 振動(Vibration)

確定申請書已充分評估包件在正常運送事件中的振動效果。並應考慮由於振動、溫度和壓力負荷的綜合應力，提供高應力系統的疲勞分析。如果封閉螺栓重複使用，驗證螺栓預緊力包含疲勞評估分析。NUREG/ CR-6007提供螺栓評估分析之指引。驗證任何會導致快速的疲勞損傷的共振振動狀態，不存在於任何包裝組件中。在包裝內部的影響應予以考慮。NUREG/ CR-2146和NUREG/ CR-0128提供

振動評估的額外指引。

3.2.5.6.6 水噴濺(Water Spray)

審查包件設計於水噴濺試驗的影響。確認本測試對材料性質無顯著影響。

3.2.5.6.7 自由墜落(Free Drop)

審查包件設計於自由墜落試驗的影響。申請書應註明試驗的因子，例如墜落方位、結合壓力、熱和冷的溫度、以及其他在第2.5.5節中討論的其它因素對自由墜落的影響。

審查自由墜落衝擊力、內部壓力、熱應力、O型環的壓縮力、螺栓預緊力，對密封蓋螺栓的綜合影響評估。螺栓的評估方法示於NUREG/ CR-6007。

審查其它包件組件評估結果，例如端口蓋(port covers)、端口蓋板(port cover plate)、和屏蔽(shield enclosures)，對於包件墜落衝擊力、內部壓力和熱應力的綜合影響。

3.2.5.6.8 角墜落(Corner Drop)

審查角墜落試驗對包件設計的影響，如果可以的話。

3.2.5.6.9 壓縮(Compression)

審查壓縮試驗對包件設計的影響，如果可以的話。

3.2.5.6.10 滲透(Penetration)

審查包件的滲透試驗的評估結果，驗證申請書已考量包件最脆弱的位置。

3.2.5.7　假設意外事故狀況下之結構評估

假設意外事故狀況下的評估，必須基於§71.73規定的試驗順序程序，為了說明及確定對包件的累積效應。評估包件承受任何一個測試的效能，必須考量，先前的試驗中造成的損傷。此外，如第2.5.6節所描述，在正常運輸條件下的試驗必須不影響包件的承受假設意外事故狀況試驗的能力。確認以上評估證明包件具有足夠的結構完整性，可滿足假設意外事故狀況下，10 CFR 71對包封容器(containment)、屏蔽和次臨界的法規要求：

- 包封容器和密封系統的非彈性變形一般是不能接受包封容器評估。
- 屏蔽組件的變形，應就屏蔽評估方面進行審查。
- 傳熱導和絕緣致組件的變形，應就熱傳評估方面進行審查。
- 次臨界影響致組件變形，應就臨界評估方面進行審查。

對於試驗的初始條件（除了浸水試驗），試驗前後的環境氣溫必須保持在-29℃(-20℉)到+38℃(100℉)之間，這對於所考慮的特徵是最不利的值。在儲存容器內的初始內壓必須是最大正常操作壓力，除非較低的內部壓力與所選擇的環境溫度是不太有利的。

3.2.5.7.1　自由墜落(Free Drop)

審查的自由墜落的評估。驗證結構完整性已被評估，包括導致嚴重的損害的墜落方位、重心、二次碰撞、傾斜方向、邊墜落、密封蓋碰撞等。一個組件最具破壞性的方

向，可能不是另一個組件最具破壞性的方向。例如像固定裝置是包件結構的一部分，它應在跌落試驗配置的選擇和墜落的方向加以考慮。對於具有鉛屏蔽的包件，鉛跌落的影響應該進行評估。NUREG/CR- 4554, Vol. 3中有鉛墜落的相關討論。

審查封蓋螺栓設計的綜合影響評估，包括：自由墜落衝擊力、內部壓力、熱應力、O型環的壓縮力、以及螺栓預緊力。螺栓的評估方法示於NUREG/ CR-6007。審查其它包件組件的評估，例如：端口蓋、端口蓋板、屏蔽等，對於包件墜落衝擊力、內部壓力和熱應力的綜合影響評估。包件組件屈曲應予以考慮。NUREG/ CR-4554，Vol. 6提供評估包封容器屈曲的相關資訊。

3.2.5.7.2 擠壓(Crush)

如果適用，審查包件的動態擠壓條件。驗證最不利方向的選擇已被說明。

3.2.5.7.3 穿刺(Puncture)

查看包件的穿刺試驗的綜合評估。驗證了其最大的傷害所預期的位置已被確定，以及是合理的。任何因自由墜落和擠壓條件而造成的損壞，必須在評估該測試時予以考慮。雖然分析方法可用於預測穿刺，但是是從層板的穿刺測試結果得到的經驗公式常用於包件的設計。專門用於包件設計開發的Nelm的公式，提供了所需用於傳統鋼-鉛-鋼鋼板層防止穿刺的最小厚度。穿刺評估方法的說明示於

NUREG/ CR-4554，Vol. 7，穿刺試驗其他注意事項示於NRC 97-02公告中。

驗證以傾斜角度、鄰近支撐組件、閥門附近的穿刺，都已經酌情考量。

3.2.5.7.4 熱傳(Thermal)

驗證該結構設計被火完全吞沒的影響，如在§71.73(c)(4)中規定。由自由墜落、擠壓、穿刺所造成的任何損壞，都必須納入包件火災試驗的初始條件。確認過程中或試驗後的包件的最大壓力，以及火災引起的燃燒或分解過程中所產生的溫度。驗證的最大熱應力，這可以在火災發生期間或在火災後，進行評估。

3.2.5.7.5 放射性物質浸沒(Immersion-Fissile Material)

如果包件內容物包括可分裂材料，則受到§71.55的法規要求，而且如果水滲入尚未假設為臨界分析，審查試樣浸入至少0.9 m(3 ft)水頭的破壞試驗結果。

3.2.5.7.6 包件浸沒(Immersion-All Packages)

單獨審查，未損壞的試樣進行水壓相當於浸沒至少15 m(50 ft)水頭（50英尺）試驗。出於測試目的，外部水壓約150 kPa(21.7 psi)被認為是能滿足這些條件。

3.2.5.8 附錄

確認附錄內包括引用的參考文獻列表、程式碼、輸入和輸出檔、試驗結果、及其他補充資訊。

如果包件是通過試驗進行評估，審查其相關說明，該說明

應包括：

- 試驗程序
- 試驗包件說明
- 試驗初始及邊界條件
- 試驗年度計畫表（計劃與實際）
- 包件組件的照片，包括任何試驗前後的結構損壞
- 試驗量測，至少包括測試包件的物理變化的文件作為試驗的結果
- 試驗結果
- 獲得這些修正結果所使用的方法

3.2.6評估結果(Evaluation findings)

安全評估報告應當包括以下類似的內容： 根據申請書聲明和陳述的審查，工作人員得出結論認為，包件的結構設計已充分描述和評估，該包件具有足夠的結構完整性，以滿足10 CFR 71的法規要求。

3.3 熱傳審查(THERMAL REVIEW)

3.3.1審查目的(Review Objective)

熱傳審查的目的是驗證該包件設計在正常運送狀況及假設意外事故狀況下的熱傳效能可以得到充分的評估，並確認包件設計符合10 CFR 71的熱傳效能的法規要求。

3.3.2審查範圍(Areas of Review)

熱傳審查應包括以下內容：

3.3.2.1 熱傳設計說明

- 設計特點
- 內容物衰變熱
- 溫度匯總表
- 密封系統最大壓力匯總表

3.3.2.2　材料性質和組件規格

- 材料熱傳性質
- 組件規格

3.3.2.3　一般注意事項

- 評定分析
- 評定試驗
- 安全範圍

3.3.2.4　正常運輸狀況下之熱傳評估

- 熱與冷
- 最大正常工作壓力
- 最大熱應力

3.3.2.5　假設意外事故狀況下之熱傳評估

- 初始條件
- 防火測試條件
- 最高溫度和壓力
- 最大熱應力

3.3.2.6　附錄

- 試驗設施說明
- 試驗結果

・適用的證明文件或規範

・詳細分析內容

3.3.3法規要求(Regulatory Requirements)

10 CFR 71適用於熱傳評估的法規要求如下：

- 包件設計必須說明和評估，以證明其滿足10 CFR 71的熱傳要求。[§71.31 (a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- 申請書必須確認用於熱傳設計的規範和標準。[§71.31(c)]
- 包件的效能必須依據10 CFR 71.71規定的正常運輸狀況和10 CFR 71.73規定的假設意外事故狀況的試驗來評估。[§71.41(a)]
- 包件必須經過設計、製造、及運輸準備，因此依據10 CFR 71.71規定的正常運輸條件試驗下，包裝的有效性不會顯著地降低。[§71.43(f)，§71.51(a)(1)]
- 包件必須經過設計、製造、及運輸準備，使表面接觸溫度不超過法規的限值。 [§71.43(g)]
- 包件設計必須在不依賴於機械冷卻系統的狀況下，滿足包封容器的法規需求。 [§71.51(c)]

3.3.4接受標準(Acceptance Criteria)

- 包件設計必須符合第3.3.3節中所列的法規要求。
- 包件必須具有足夠的熱傳效能，以滿足10 CFR 71在正常運輸狀況和假設意外事故狀況下對包封容器、屏蔽、次臨界和溫度的法規要求，。

3.3.5審查程序(Review Procedures)

熱傳審查應確保包件設計已經詳細及充分的描述和評估，在正

常運輸狀況和假設意外事故狀況下的熱測試，並且符合10 CFR 71熱傳效能的法規要求。熱傳審查是基於申請書一般資訊和結構評估的描述和評估。同樣地，熱傳評估的審查結果與申請書其他章節也將一併考量。熱傳審查與其他章節審查相對之資訊架構如圖3-3所示。

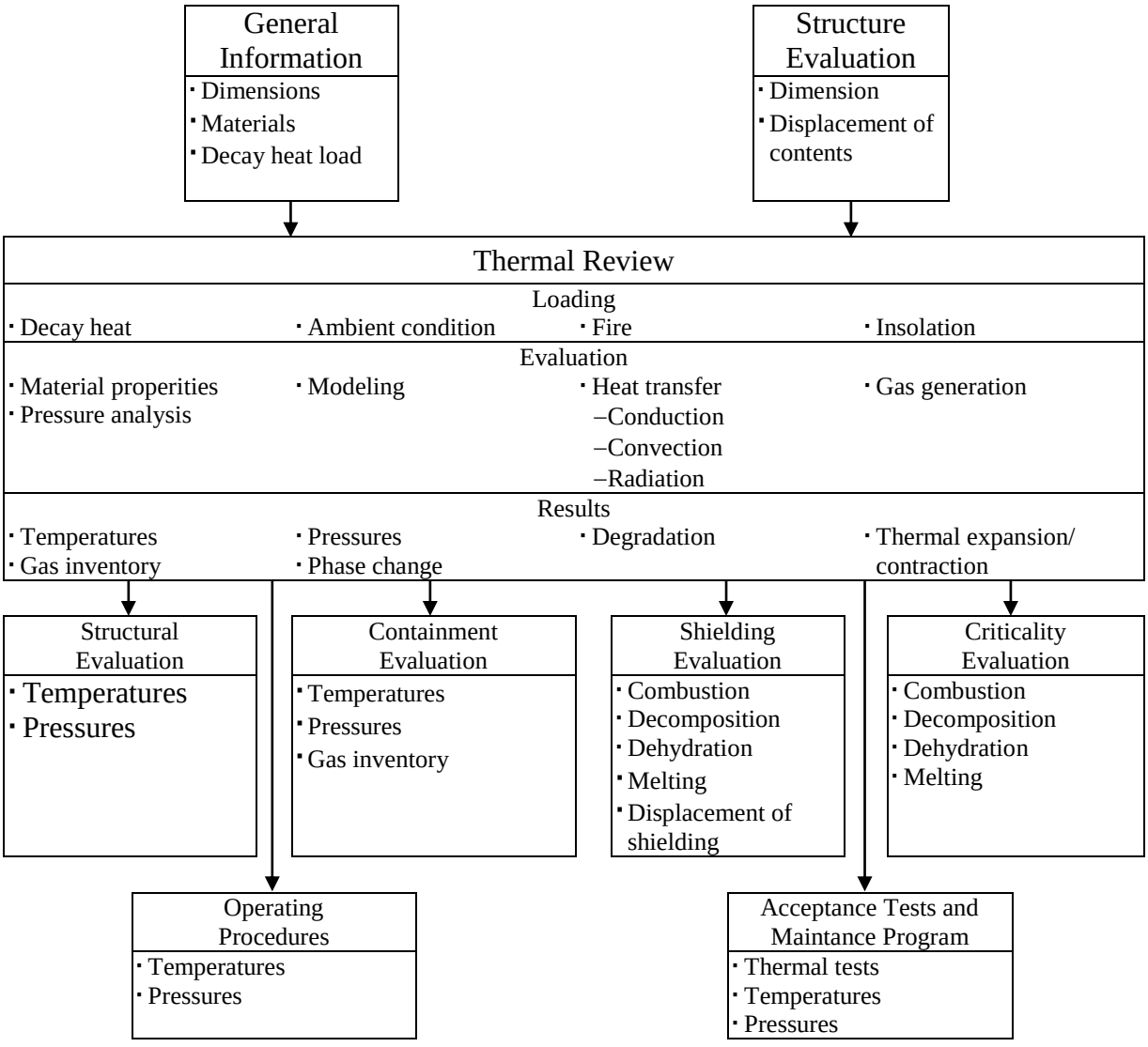


圖3-1 熱傳審查與各章節審查對應之資訊架構

3.3.5.1　熱傳設計說明

熱傳設計應說明包括以下項目：

- 設計特性
- 內容物衰變熱
- 溫度匯總表
- 包封容器之最大壓力匯總表

3.3.5.2　材料性質和組件規格

應包括以下項目：

- 材料熱傳性質
- 組件規格

3.3.5.3　一般注意事項

應包括以下項目：

- 評定分析
- 評定試驗
- 安全範圍

3.3.5.4　正常運輸狀況下之熱傳評估

- 熱與冷
- 最大正常工作壓力
- 最大熱應力

3.3.5.5　假設意外事故狀況下之熱傳評估

- 初始條件
- 防火測試條件
- 最高溫度和壓力

- 最大熱應力

3.3.5.6 附錄

- 試驗設施說明
- 試驗結果
- 適用的證明文件或規範
- 詳細分析內容

## 3.4 包封容器審查(CONTAINMENT REVIEW)

### 3.4.1審查目的(Review Objective)

審查的目的驗證該包件的設計滿足10 CFR 71，正常運輸狀況和假設意外事故狀況下，包封容器的法規要求。

### 3.4.2審查範圍(Areas of Review)

審查範圍必須說明包封容器相關資訊，包括包封容器邊界及有關放射性物質之特殊法規要求；一般注意事項，包括甲型包件或乙型包件，以及易燃性氣體之產生等；乙型包件正常運輸狀況假設意外事故狀況下之密封設計標準，洩漏率試驗等。

### 3.4.3法規要求(Regulatory Requirements)

10 CFR 71適用於包封容器審查的法規要求如下：

- 包件設計必須詳細及充分描述和評估，以證明其符合10 CFR 71對包封容器的法規要求。[§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- 申請書必須確認適用於包封容器設計的規範和標準。[§71.31(c)]
- 包件必須具備包封容器系統，不能被無意開啟，或被包件內提高

的壓力撐開的鎖緊裝置，安全地緊閉。[§71.43(c)]

- 包件必須由保證不會有顯著化學、電化學、或其它反應的材料和構造所製造。 [§71.43(d)]
- 包件上的任何閥門或類似的裝置必須被保護，以防止未經授權的操作，且除了卸壓閥，必須用密封蓋來防止任何洩漏。 [§71.43(e)]
- 包件必須經過設計、施工、裝運準備，以保證依據§71.71規定在正常運送狀況下的試驗，沒有任何放射性物質的損失或散佈。 [§71.43(f), §71.51(a)(1)]
- 包件不包含在運輸過程中連續排氣(venting)的功能。 [§71.43(h)]
- 乙型包件必須符合§71.51(a)(1)包封容器在正常運輸狀況和§71.51(a)(2)假設意外事故狀況下，不依靠過濾器或機械冷卻系統的法規要求。 [§71.51]
- 含有超過0.74TBq(20 Ci)鈽的包件，必須滿足鈽的特殊包封容器的法規要求。[§71.63]

3.4.4接受標準(Acceptance Criteria)

- 包件必須滿足第3.4.3節中列出的法規要求。
- 包件設計必須符合10 CFR 71正常運輸狀狀和假設意外事故狀況下的包封容器法規要求。

3.4.5審查程序(Review Procedures)

包封容器審查應確保包件設計已詳細及充分的描述和評估，以滿足10 CFR 71的正常運輸狀況和假設意外事故狀況下的包封容器的法規要求。包封容器(containment)的審查是基於申請書中一般資訊、結構評估、熱傳評估中的說明和評估進行。同樣，包封容器審

查結果會在操作程序、驗收測試和維護計畫的審查中考量。包封容器審查與其他章節審查的資訊架構示於圖3-4。

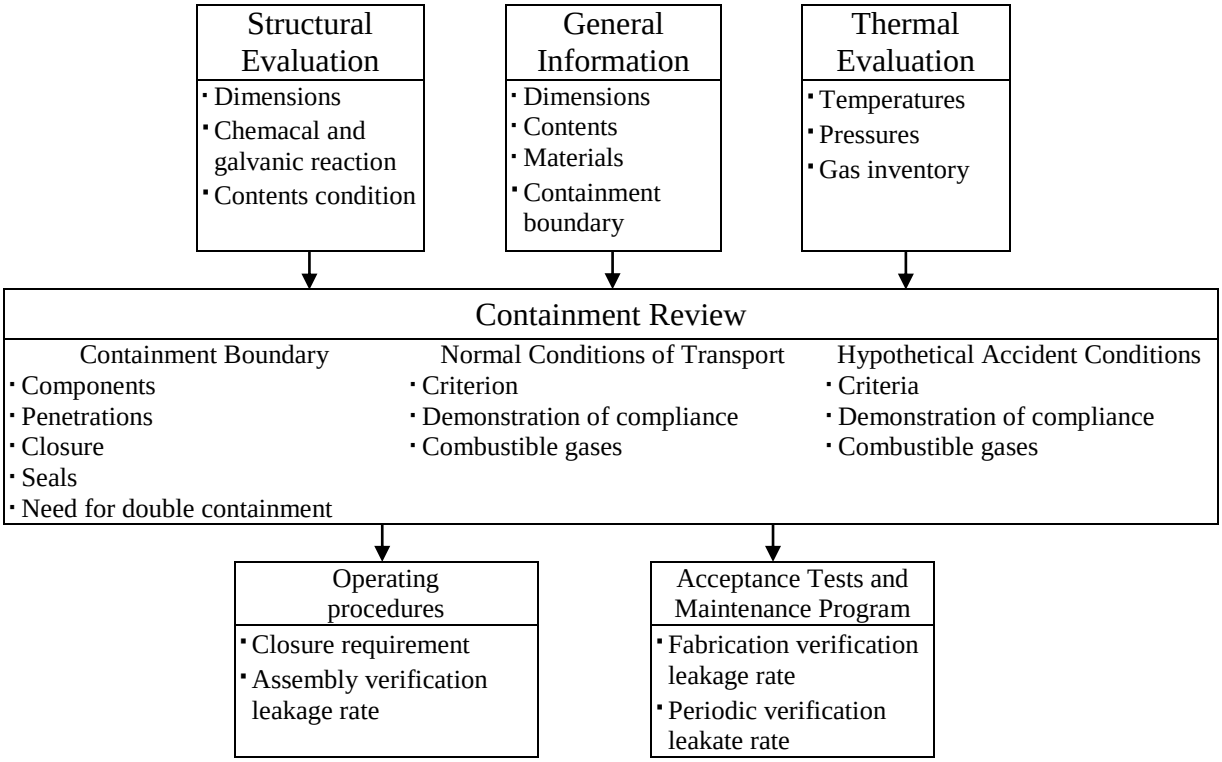


圖3-4 包封容器審查與其他章節審查的資訊架構

## 3.5 屏蔽審查(SHIELDING REVIEW)

### 3.5.1審查目的(Review Objective)

屏蔽審查的目的是驗證該包件設計滿足10 CFR 71在正常運輸狀況和假設意外事故狀況下對外部輻射的法規要求。

### 3.5.2審查範圍(Areas of Review)

屏蔽設計的描述和評估應進行審查，包括以下內容：

#### 3.5.2.1 屏蔽設計說明

- 設計特性功能

- 最大輻射分級匯總表

3.5.2.2 放射源(Radiation Source)

- 伽瑪源
- 中子源

3.5.2.3 屏蔽模式(Shield model)

- 放射源和屏蔽的配置
- 材料性質

3.5.2.4 屏蔽評估

- 方法
- 輸入和輸出數據
- 通量-劑量率轉換
- 外部輻射程度

3.5.2.5 附錄

3.5.3法規要求(Regulatory Requirements)

適用於屏蔽審查的10 CFR 71法規要求如下所示：

- 包件設計必須詳細及充分的描述和評價，以證明其符合10 CFR 71對屏蔽的法規要求。
- 申請書必須確認用於屏蔽設計的現有規範和標準。[§71.31(c)]
- 包件必須經過設計、施工、及運送準備，在符合10 CFR 71.71正常運輸狀況的法規試驗下，使得外部輻射不會顯著增加。[§71.43(f), §71.51(a)(1)]
- 依據10 CFR 71.71正常運輸狀況下規定的試驗，外部輻射必須符合§71.47(a)非專用，或§71.47(b)專用(exclusive use)交運的法規

要求。[§71.47]

- 依據§71.73（假設意外事故狀況）規定的試驗，外部輻射從包件表面一公尺處量測不得超過10 mSv/h(1 rem/h)。[§71.51(a)(2)]

3.5.4接受標準(Acceptance Criteria)

- 包件必須滿足第3.5.3節所列的法規要求。
- 包件設計必須符合10 CFR 71正常運輸狀況及假設意外事故狀況下外部輻射之法規要求。

3.5.5審查程序(Review Procedures)

屏蔽審查應確保包件設計已經詳細的描述和評估，以滿足10 CFR 71正常運輸狀況和假設意外事故狀況下外部輻射的法規要求。屏蔽審查是基於一般資訊、結構評估和熱傳評估章節中的說明和評估。屏蔽審查的結果在操作程序、驗收測試和維護計畫的審查中被考量。屏蔽審查和其他章節審查的資訊架構示於圖3-5。

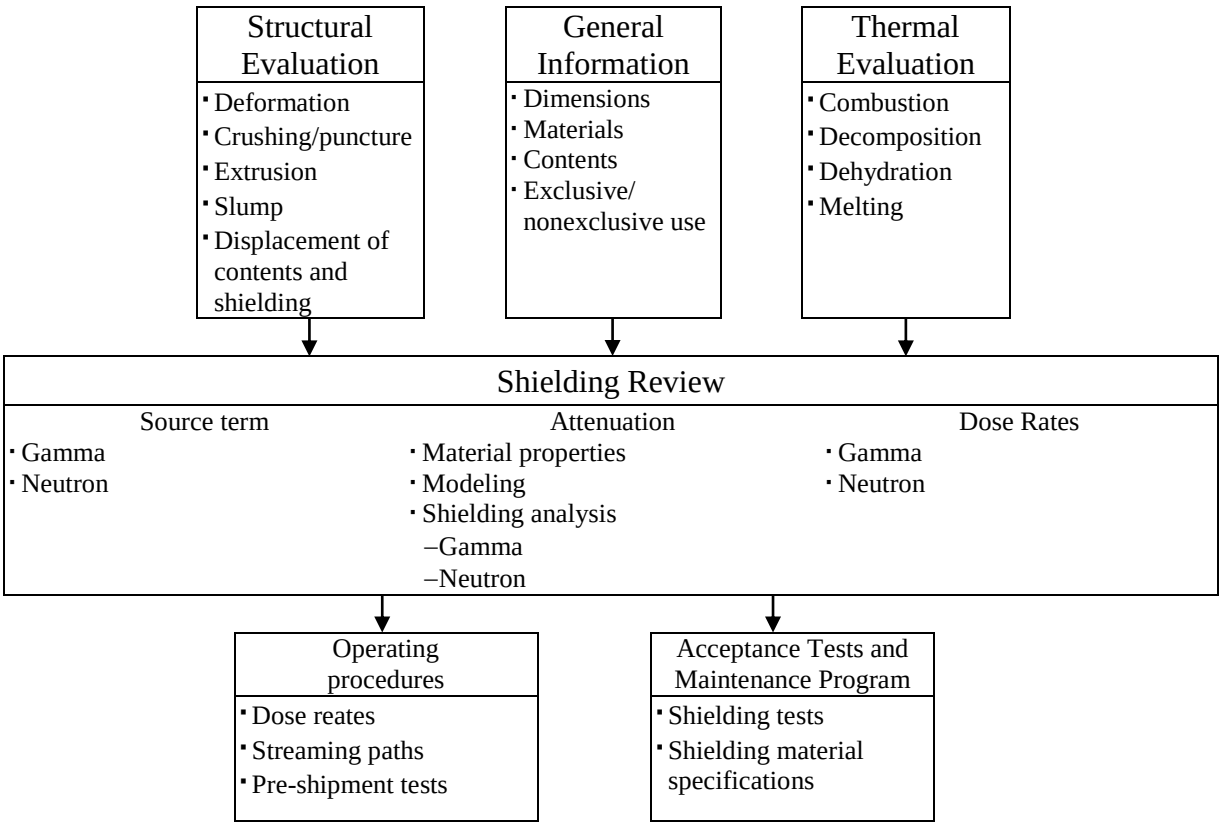


圖5-1 屏蔽審查和其他章節審查的資訊架構

## 3.6 臨界審查(CRITICALITY REVIEW)

### 3.6.1審查目的(Review Objective)

審查的目的是驗證該包件的設計符合10 CFR 71正常運輸狀況和假設意外事故狀況下的臨界安全要求。

### 3.6.2審查範圍(Areas of Review)

臨界設計說明及評估應被審查，臨界審查應包含下列各項：

#### 3.6.2.1 臨界設計說明

- 設計特徵

- 臨界評估匯總表
- 運送指數

3.6.2.2 可分裂物質

3.6.2.3 一般注意事項

- Model Configuration
- 材料性質
- 程式碼和截面稽資料庫
- 最大活度的示範
- 驗證分析

3.6.2.4 單一包件評估

- 配置
- 結果

3.6.2.5 正常運輸狀況下包件陣列評估

- 配置
- 結果

3.6.2.6 假設意外事故狀況下包件陣列評估

- 配置
- 結果

3.6.2.7 評估基準

- 基準實驗的適用性
- 偏差測定

3.6.3法規要求(Regulatory Requirements)

適用於可分裂物質臨界審查的10 CFR 71的法規要求如下：

- 包件設計必須詳細及充分的描述和評價，以證明其符合10 CFR 71臨界之法規要求。[§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- 申請書必須確認適用於臨界設計的規範和標準。[§71.31(c)]
- 可分裂物質的未知性質必須假定為會導致最高的中子增殖。[§71.83]
- 包件必須經過設計、製造、和裝運準備，這樣在§71.71規定的正常運輸狀況下試驗中，以確保不會有顯著減少包件的有效性。[§71.43(f), §71.51(a)(1), §71.55(d)(4)]
- 單一包件必須符合§71.55(d)在正常運輸條件下的法規要求。
- 單一包件必須符合§71.55(e)在假設意外事故狀況下的法規要求。
- 單一包件必須經過設計和製造，並限制內容物，使得如果水洩漏到包封容器內時，仍可以維持次臨界。[§71.55(b)]
- 包件陣列在正常運輸狀況和假設意外事故條件下，仍必須維持次臨界。 [§71.59(a)]
- 可分裂材料包件必須指定核臨界控制運輸指數，以限制單次交運包件的數量。 [§71.59，§71.35(b)]

3.6.4接受標準(Acceptance Criteria)

- 包件必須滿足第3.6.3節的法規要求。
- 包件設計必須符合10 CFR 71在正常運輸狀況及假設意外事故狀況下之臨界安全要求。

3.6.5審查程序(Review Procedures)

臨界審查應確保包件設計已經詳細及充分的描述和評估，以滿足10 CFR 71在正常運輸狀況和假設意外事故狀況下的臨界安全的要求。臨界審查是基於在一般資訊、結構評估、以及熱傳評估章節的說明和評估。臨界評估的結果，在操作程序、驗收測試和維護計畫的審查中被考量。臨界審查與其他章節對應的資訊結構示於圖3-6。

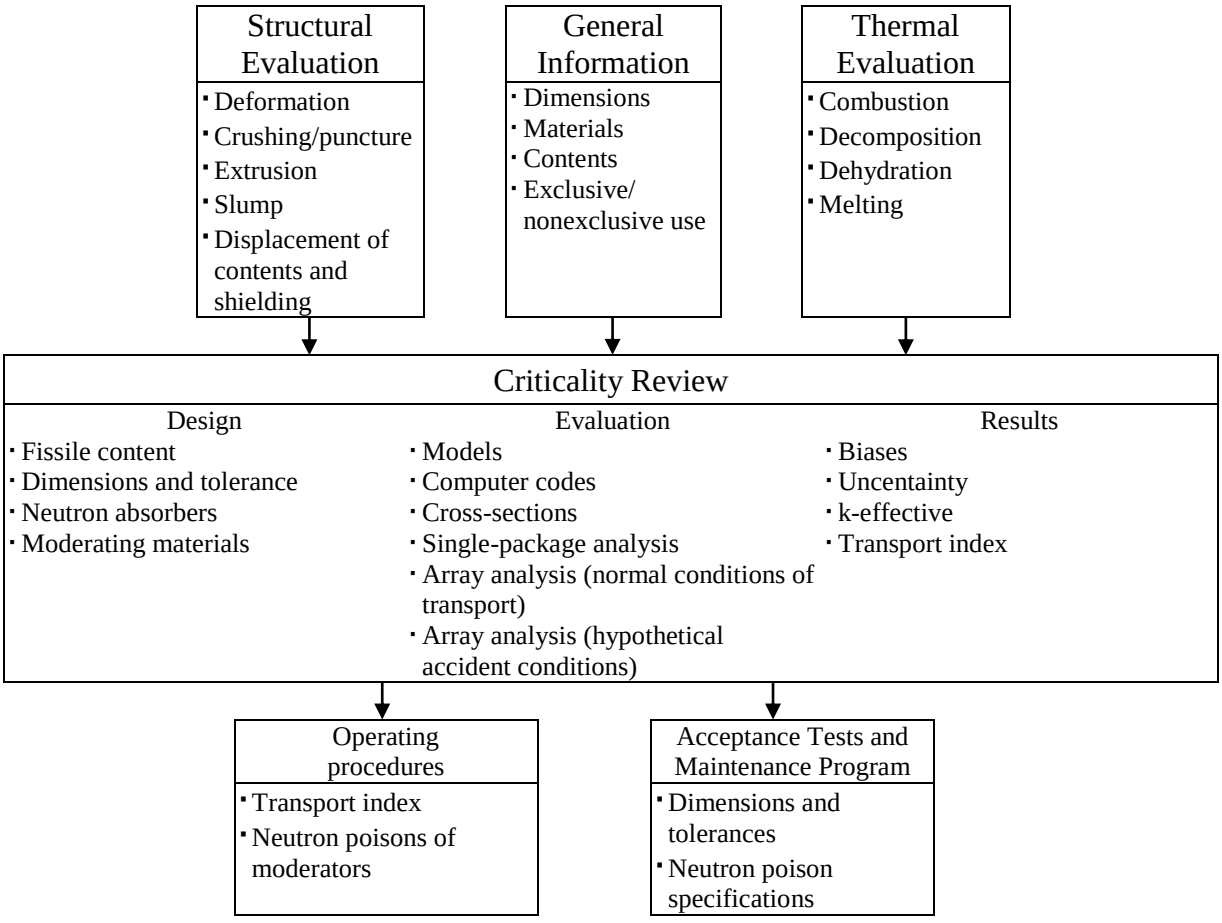


圖3-6 臨界審查與其他章節對應的資訊結構

3.6.6評估結果(Evaluation findings)

安全評估報告應該要有以下類似之評估結果：根據審查申請書的聲明和陳述中，工作人員的結論是，核臨界安全設計已充分說明、評估，包件符合10 CFR 71的次臨界法規要求。

## 3.7 操作程序審查(OPERATING PROCEDURES REVIEW)

3.7.1審查目的(Review Objective)

審查目的是驗證操作控制及程序符合10 CFR 71的法規要求，以及確認包件將依批准的操作程序進行操作。

3.7.2審查範圍(Areas of Review)

操作程序必須經過審查，以確保包件的使用操作與批准的程序一致。操作程序審查應包括以下內容：

3.7.2.1 包件裝載

- 裝載準備
- 裝載內容物
- 運輸準備

3.7.2.2 包件卸載

- 從載具接收包件
- 移除內容物

3.7.2.3 空包件的運輸準備

3.7.2.4 其他程序

3.7.2.5 附錄

3.7.3法規要求(Regulatory Requirements)

適用於操作程序審查的10 CFR 71法規要求如下所示：

- 申請書必須確認用於操作程序制定的規範和標準。[§71.31(c)]
- 可分裂材料裝運申請必須包括：對運輸、裝載、卸載及吊運等作業，預先規劃可能特殊控制和預防措施，以及針對防止發生意外事故或延遲採取的任何特殊控制。[§71.35(c)]
- 因為輻射活度較高而要求專用交運的包件，必須針對載具提供書面說明以進行掌控。[§71.47(b-d)]
- 每次交運前，持照者必須確認包件符合10 CFR 71例行測定(routine-determination)的法規要求[§71.87]
- 在包件送上載具前，持照者必須將如何安全地打開包件的特別說明，寄給收貨人使用，以符合10 CFR20.1906(e)的規定。 [§71.89]

3.7.4接受標準(Acceptance Criteria)

操作程序必須符合第3.7.3節的法規要求。操作程序必須確認該包件的操作方式與安全評估報告基礎一致的方式來操作。

3.7.5審查程序(Review Procedures)

審查應確認操作控制及程序符合10 CFR 71的要求，而這些程序足以保證包件將與批准的一致方式來操作，而不需要另外再提出更詳細的程序。在申請書中的操作步驟中指定的承諾一般包括：品質符合證書作為包件核准的證明。包件操作和準備交運必須按照詳細的書面程序進行。提交及運行的程序應建立最低限度的步驟，操作程序是提交的申請書的一部份，應建立最低限度必要的步驟，以保證在正常運輸狀況和假設意外事故狀況下包件的安全性。這些步

驟應盡量依照順序提出。操作程序審查是基於在申請書中一般資訊、結構評估、熱傳評估、包封容器(Containment)、屏蔽評估、和臨界評估章節中提供的說明和評估執行。操作程序審查的結果被認為是驗收測試和維護方案的審查。操作程序審查與其他章節審查的資訊結構如圖7-1所示。

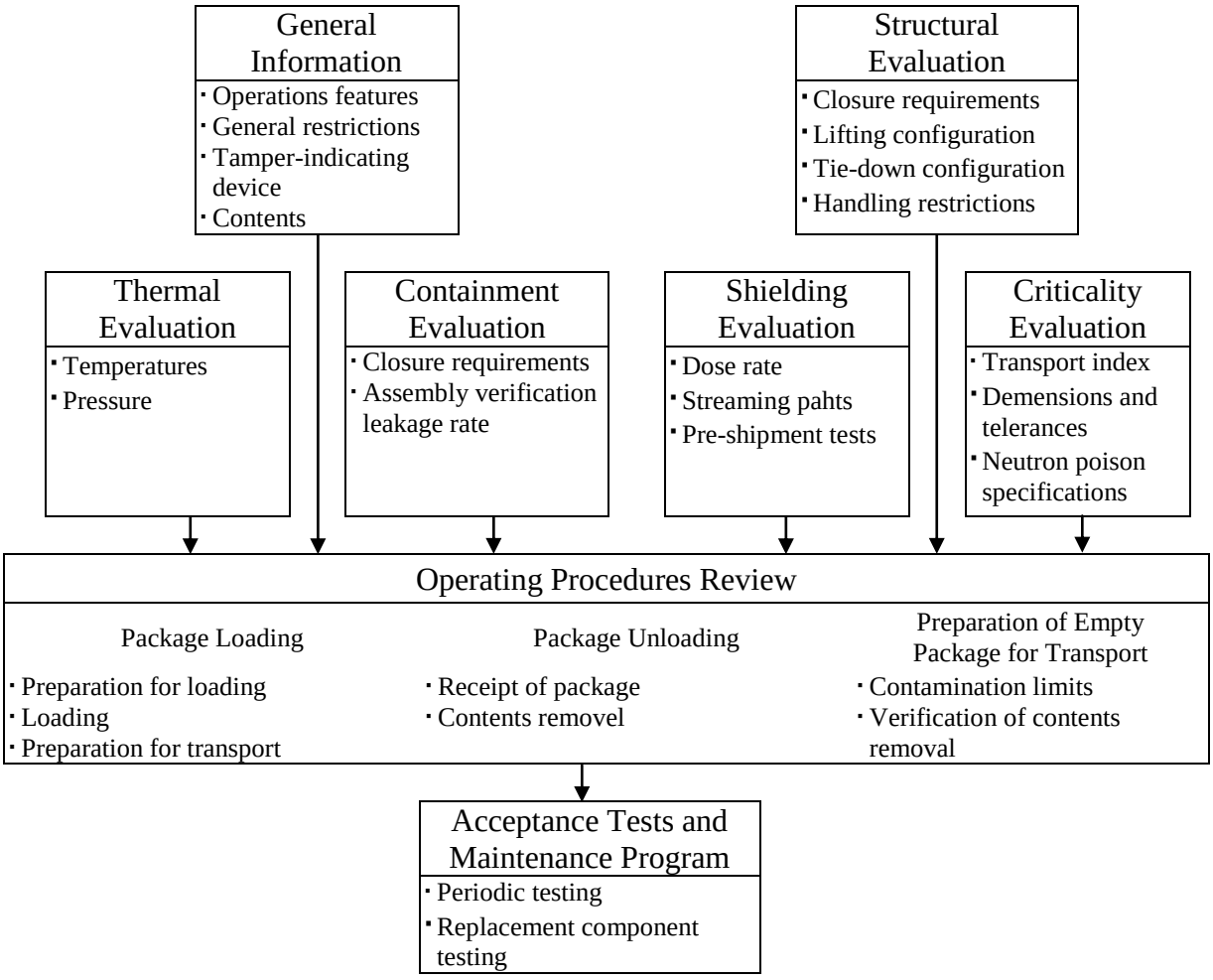


圖3-7 操作程序審查與其他章節審查的資訊結構

3.7.6評估結果(Evaluation findings)

安全評估報告應該包含類似以下結果：根據審查申請書的聲明和陳述中，工作人員的結論是，操作程序已充分說明、評估，確認包件符合10 CFR 71的法規要求。

## 3.8 驗收試驗和維護方案審查(ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW)

3.8.1審查目的(Review Objective)

審查的目的是驗證驗收測試的包件滿足10 CFR 71的維護方案，以保證使用壽命期間的包件性能的法規要求。

3.8.2審查範圍(Areas of Review)

驗收測試和維護方案應進行審查，審查應包括下列各項：

3.8.2.1 驗收試驗

- 目視檢查和測量
- 焊接檢驗
- 結構和壓力測試
- 洩漏測試
- 組件與材料測試
- 屏蔽測試
- 熱傳試驗

3.8.2.2 維護方案

- 結構和壓力測試
- 洩漏測試

- 組件與材料測試
- 熱傳試驗
- 雜項測試

3.8.2.3 附錄

3.8.3法規要求(Regulatory Requirements)

10 CFR 71適用於驗收測試和維修方案的審查審查要求如下：

3.8.3.1 驗收試驗

- 申請書必須確認適用的規範、標準和用於包件的驗收測試的品質保證的規定。[§71.31(c), §71.37(b)]
- 第一次使用前，每個包裝的製造必須驗證是按照核准的設計。[§71.85(c)]
- 第一次使用之前，每個包裝必須檢查是否有顯著降低其有效性的裂紋、針孔、未受控制的孔隙、或其它缺陷。 [§71.85(a)]
- 第一次使用之前，如果包件的最大正常操作表計壓力超過35 kPa(5 psi)，每個包裝的包封容器系統，必須在內部壓力超過最大正常操作壓力至少50%的測試，以驗證其能力，保持在該壓力下的結構完整性。 [§71.85(b)]
- 第一次使用前，每個包裝上必須顯著而耐久地標有它的型號、序號、毛重，以及由NCR指定的包件識別碼。 [§71.85(c)]
- 持照者必須履行NRC認為適當的任何測試。 [§71.93(b)]

3.8.3.2 維護方案

- 申請書必須確認適用的規範、標準和用於包件的驗收測試的品質保證的規定。[§71.31(c), §71.37(b)]

- 包裝除了淺表面的缺陷，如標識或凹痕等，必須被保持在未受損傷的狀況。 [§71.87(b)]
- 任何緩和劑或中子吸收劑存在可分裂材料包件中，如果需要的話，在每次交運之前都必須驗證。 [§71.87(g)]
- 持照者必須履行任何NCR認為適當的測試。 [§71.93(b)]

3.8.4接受標準(Acceptance Criteria)

- 驗收測試和維護方案必須符合第3.8.3節的法規要求。
- 第一次使用前，每個包裝必須受到適當的驗收測試，以驗證它已經按照經核准的設計製造，而且其性能將滿足10 CFR 71的法規要求。
- 維護方案必須足以確保該包件將可符合在其預期的使用壽命。

3.8.5審查程序(Review Procedures)

審查應確保針對指定的包件有適當的驗收測試和維修方案。驗收測試和維修方案的審查是基於對申請書的前面幾個章節的描述和評估結果。驗收測試和維護方案審查與其他章節的資訊架構如圖3-8所示。在申請書中驗收測試和維護程序的承諾通常包括包件核准條件的品質符合證明書。

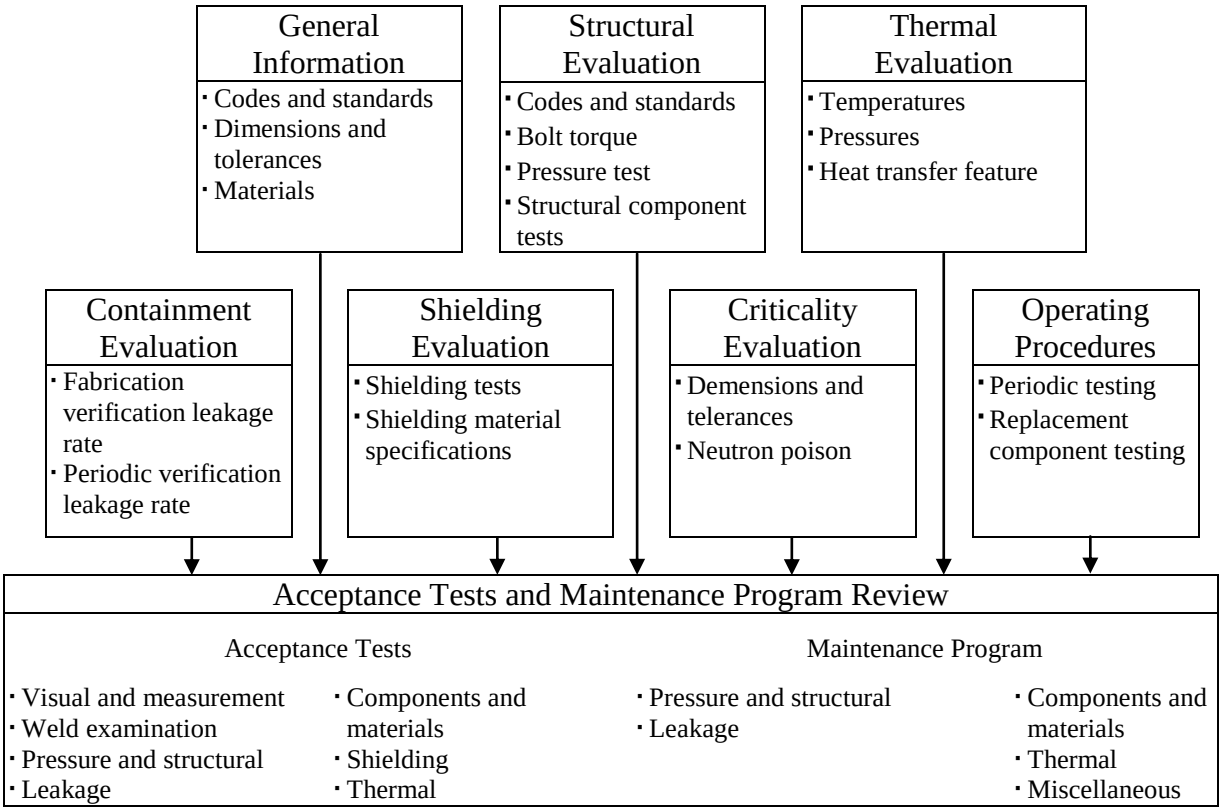


圖3-8 驗收測試和維護方案審查與其他章節的資訊架構

### 3.8.6評估結果(Evaluation findings)

安全評估報告應該包含類似以下結果：根據審查申請書的聲明和陳述中，工作人員的結論是，包裝驗收試驗已充分說明、評估，確認包件符合10 CFR 71的法規要求，維護方案足以確保該包件將可符合在其預期的使用壽命。

# 4. 結論與建議

除役廢棄物分類主要仍以低放射性 A 類、B 類、C 類，以及超 C 類廢棄物為主，在提出盛裝容器申請時應詳細說明其目的及策略，包括是用於貯存、長期貯存、運送、最終處置，或是具有多重使用目的，以提供管制機關就其目的及策略進行審查。

由於放射性廢棄物盛裝容器及包件因不同目的及效能，具有極大的差異性，以及有許多不同的方法可以評估盛裝容器的設計，所以沒有任何審查計畫可以詳細的包括每一種申請狀況，因此需要修改及擴充審查計畫的相關指引來適應特定的盛裝容器設計，不同的盛裝容器設計及使用目的，適用不同的法規及規範。

最後的評估結果應該要於安全評估報告清楚載明，依據相關的法規及規範，經過詳細及充分的評估所得到的結論是，包括一般資訊、結構、熱傳、包封容器、屏蔽、臨界、操作程序、驗收試驗和維護方案皆已充分說明、評估，確認包件符合相關法規要求，足以確保該包件符合其預期的使用壽命。

# 5.參考文獻


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# 附錄

A-1 放射性照相廢棄物盛裝容器(RADIOGRAPHY PACKAGES)

A-2 乙型廢棄物盛裝容器(TYPE B WASTE PACKAGES)

A-3 未照射燃料盛裝容器(UNIRRADIATED FUEL PACKAGES)

A-4 低濃縮氧化鈾盛裝容器(LOW ENRICHED URANIUM OXIDE PACKAGES)

A-5 超鈾廢棄物盛裝容器(TRANSURANIC WASTE PACKAGES)

A-6 低濃縮六氟化鈾盛裝容器(LOW ENRICHED URANIUM HEXAFLUORIDE PACKAGES)

A-7 高濃縮鈾或鈽盛裝容器(HIGH ENRICHED URANIUM OR PLUTONIUM PACKAGES)

A-8 乙型特殊型式盛裝容器(TYPE B SPECIAL FORM PACKAGES)

# APPENDIX A1:

# RADIOGRAPHY PACKAGES

## A1.1 Package Type

### A1.1.1 Purpose of Package

These packages include radiographic exposure devices and radiographic source changers. The purpose of an exposure device is to transport a Type B quantity of special form radioactive material for use as a radiographic gamma source. The purpose of the source changer device is to transport a radiographic gamma source to and from an exposure device and to exchange radiographic sources with that exposure device.

### A1.1.2 Description of a Typical Package

A typical packaging used as an exposure device consists of a lead or depleted uranium shield inside a welded steel or titanium housing. The shield includes a metallic S-shaped tube that houses the source during transport and allows movement of the source into position for radiography. The shield may be fixed in position by retention cups welded to end plates of the housing and by foam between the shield and the housing.

The source is attached to the end of a short metallic cable, or pigtail. A securing lock mechanism is installed at one end of the housing to maintain the source in a fixed position during transport. A safety plug assembly installed at the other end of the S-tube provides a redundant mechanism to prevent movement of the source toward an outlet.

The content of a package used as an exposure device is one radiographic gamma source ($^{60}$Co or $^{192}$Ir) in Type B special form.

The package is typically hand-carried by one person using a handle attached to the housing, although it is sometimes mounted on wheels.

A typical packaging used as a radiographic source changer is similar to that used as an exposure device. A source changer may contain multiple sources, typically housed in U-shaped tubes. In addition to its function as a transportation package, a source changer is used to move sources either from or to an exposure device. Although the remainder of this appendix specifically addresses exposure devices, the review of a source changer is similar.

A sketch of a typical radiographic exposure device is presented in Figure A1-1.

## A1.2 Package Safety

### A1.2.1 Safety Functions

The principal safety function of these packages is to retain the radiographic source and to provide gamma shielding. Containment is provided primarily by the special form source itself. These packages do not contain fissile material.

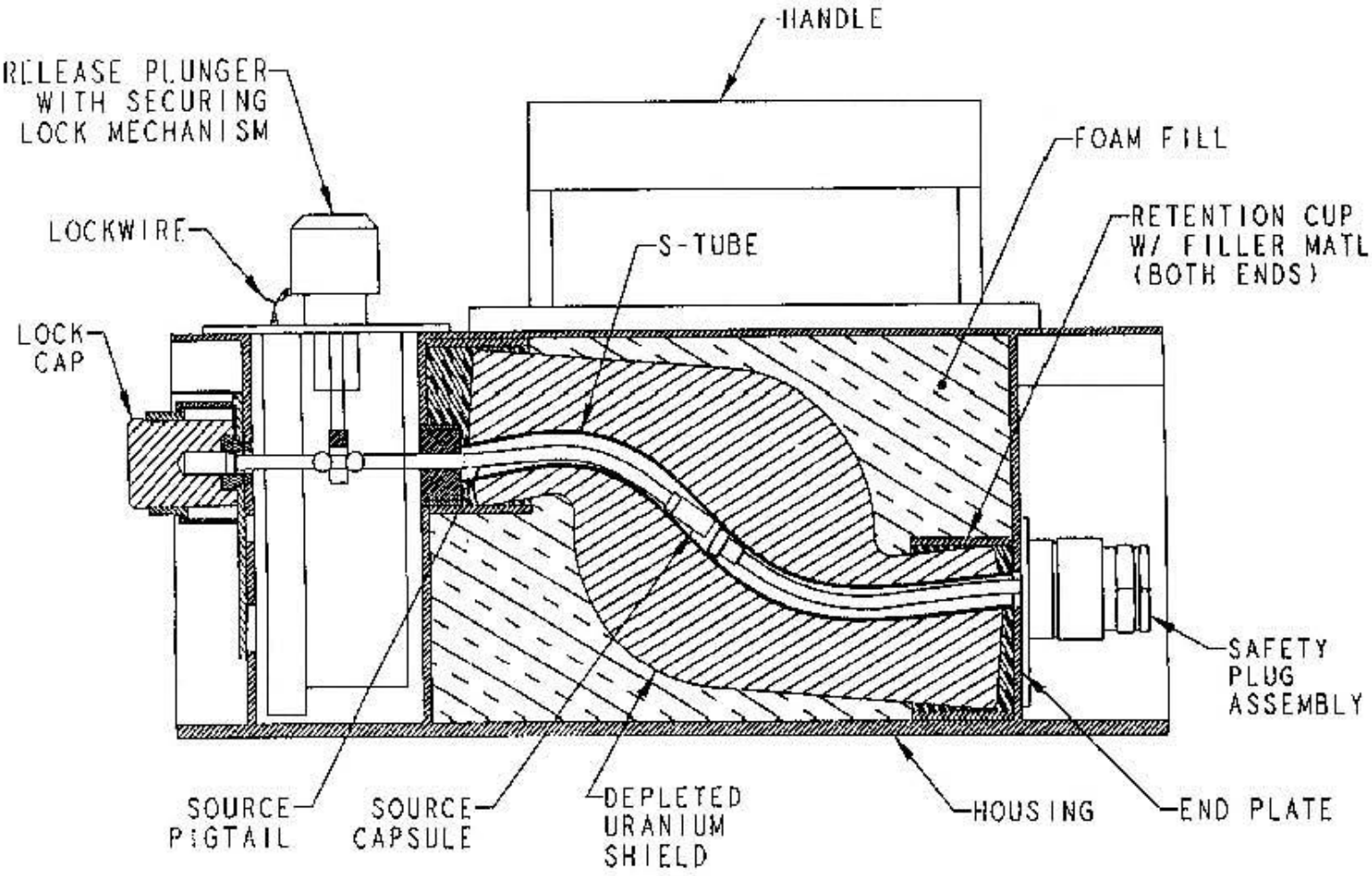


FIGURE A1-1
RADIOGRAPHIC EXPOSURE DEVICE PACKAGE

### A1.2.2 Safety Features

- A lead or depleted uranium shield provides gamma shielding.
- A securing lock mechanism positions the source pigtail within the S-tube in the shield during transport to prevent high radiation fields and radiation streaming.
- A safety plug assembly at the opposite end of the tube provides a redundant mechanism to prevent movement of the source.
- The housing, foam, and other structural materials protect the shield and S-tube from damage.

### A1.2.3 Typical Areas of Review for Package Drawings

- Housing features, including dimensions, material, thickness, and welds
- Foam material and density
- Shield dimensions and material, including supplemental shielding, its weight, dimensions, and method of attachment
- Material, wall thickness and curvature of S- or U-tube
- Lock mechanism specifications
- Other structural features, including bolts, pins, and retention cups, as applicable.

### A1.2.4 Typical Areas of Safety Review

- The general information review verifies that the contents are restricted to special form and that the source nuclide and maximum allowable activity are specified. Specification of content activity is typically expressed as "Bq (output)" or "Ci (output)" to denote that the activity is determined in accordance with ANSI N432-1980.
- The structural and thermal reviews evaluate the ability of the shield to perform its intended function under normal conditions of transport and hypothetical accident conditions. These reviews address:
  - Damage to the shielding
  - Misalignment of the S-tube
  - Damage to the S-tube resulting in exposure of the depleted uranium shield and possible oxidation of the uranium or eutectic reaction between the uranium and other package components
  - Damage to the securing lock mechanism
  - Movement of the source relative to the shielding.
- The shielding review evaluates the ability of the package to satisfy the maximum allowable external radiation levels under normal conditions of transport and hypothetical accident conditions. Shielding requirements are often demonstrated by measuring the dose rates from a

gamma test source and scaling them according to the maximum allowed activity of the contents. Key issues include:

- Ensuring that the locations of the maximum radiation levels on the surface of the package, including near the ends of the S-shaped source tube, and at one meter from the surface have been identified
- Determining that the size (active depth and diameter) of the detector is appropriate for providing dose rate measurements at the regulatory locations (because of the small size of the package, corrections may be needed to account for the size of the detector probe volume) (ANSI N43.9-1991)
- Examining the design of the source assembly and securing lock mechanism, including pigtail and locking balls. A small movement in source position can result in a significant increase in external radiation levels
- Verifying that no significant increase in radiation occurs as a result of the tests for normal conditions of transport
- Confirming that the radiation levels under normal conditions of transport and hypothetical accident conditions are satisfied.

- The review of operating procedures confirms that the source is securely locked in position before shipment. This review also evaluates procedures to verify by physical means that the source has been removed before shipment of an "empty" package. Because of shielding effectiveness and radiation from uranium shielding itself, verification by radiation measurements alone may not be sufficient. The procedure should be capable of detecting remaining sources if the pigtail is clipped off.
- The review of acceptance tests and maintenance program verifies that appropriate fabrication and periodic verification tests are performed to demonstrate effectiveness of the shielding. The review also verifies that appropriate inspections are performed to monitor any wearing of the S-tube.

Several NRC Information Notices (85-07, 87-47, 88-18, 88-33, 90-24, 90-35, 90-82) provide additional detail on safety issues relevant to the transport of radiography packages.

## References

American National Standards Institute, ANSI N43.9-1991, "American National Standard for Gamma Radiography—Specifications for Design and Testing of Apparatus," New York.

National Bureau of Standards, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography," ANSI N432-1980, Washington, DC.

U.S. Nuclear Regulatory Commission, "Contaminated Radiography Source Shipments," NMSS Information Notice 85-07, January 29, 1985.

U.S. Nuclear Regulatory Commission, "Malfunction of Lockbox on Radiography Device," NMSS Information Notice 88-18, April 25, 1988.

U.S. Nuclear Regulatory Commission, "Recent Problems Involving the Model SPEC 2-T Radiographic Exposure Device," NMSS Information Notice 88-33, May 27, 1988.

U.S. Nuclear Regulatory Commission, "Requirements for Use of Nuclear Regulatory Commission-(NRC-) Approved Transport Packages for Shipment of Type A Quantities of Radioactive Material," NMSS Information Notice 90-82, December 31, 1990.

U.S. Nuclear Regulatory Commission, "Transportation of Model SPEC 2-T Radiographic Exposure Device," NMSS Information Notice 90-24, April 10, 1990.

U.S. Nuclear Regulatory Commission, "Transportation of Radiography Devices SSINS No. 6835," NMSS Information Notice 87-47, October 5, 1987.

U.S. Nuclear Regulatory Commission, "Transportation of Type A Quantities of Radioactive Materials," NMSS Information Notice 90-35, May 24, 1990.

# APPENDIX A2:

# TYPE B WASTE PACKAGES

## A2.1 Package Type

### A2.1.1 Purpose of Package

The purpose of this type of package is to transport a Type B quantity of dry, radioactive, irradiated, and contaminated solid materials.

### A2.1.2 Description of a Typical Package

A typical packaging consists of a steel-encased, lead-shielded cylinder with impact limiters attached at both ends. The packaging may be protected by a thermal shield, consisting of a thin metal shell separated from the lead-filled cylinder by a wire wrap. Closure is provided by a bolted steel lid, which may also include lead shielding. Two concentric O-rings are installed in grooves typically on the underside of the lid. The lid includes a leak test port between the O-rings and sometimes a vent port. The bottom of the packaging contains a sealed drain port.

A typical packaging may be sized to transport ion-exchange resins, process solids, or irradiated hardware, such as control rod blades. It is approximately 3.3 m (about 11 ft.) in length and 1.3 m (about 4 ft.) in diameter (without impact limiters) and can weigh as much as 35 tons (without contents). The packaging generally has two or four trunnions near the top for lifting, and two near the bottom for rotation.

The contents of the package consist of a Type B quantity of dry, radioactive, irradiated and contaminated solid materials, generally within a secondary container. The maximum content weight may approach five tons, including shoring. The radioactive contents typically include waste containing mixed fission products and activation products. The fissile material content of these packages is limited to that permitted by the 10 CFR Part 71 general license provisions for fissile material packages, or exempt quantities.

A sketch of a typical Type B waste package is presented in Figure A2-1.

## A2.2 Package Safety

### A2.2.1 Safety Functions

The principal safety function of the package is to provide gamma shielding and containment.

### A2.2.2 Safety Features

- The lead shield provides gamma shielding. The neutron source is not significant.
- The inner vessel provides containment of the radioactive material. Although secondary containers are often used, they do not provide a containment function.

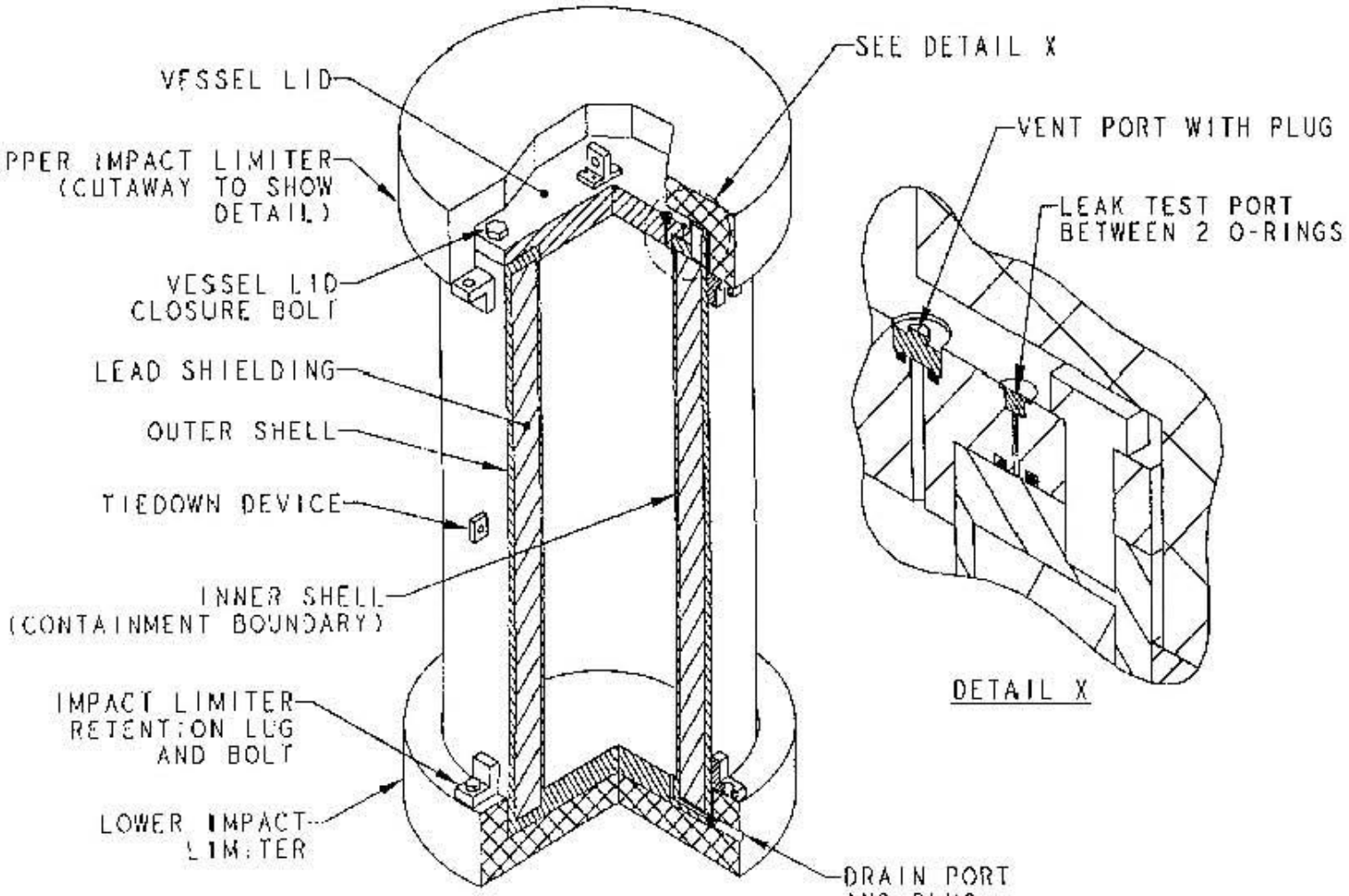


FIGURE A2-1
TYPE B WASTE PACKAGE

### A2.2.3 Typical Areas of Review for Package Drawings

- Containment vessel body
  - Materials of construction
  - Dimensions and tolerances of structural shell and shielding material
  - Fabrication codes or standards
  - Weld specifications, including codes or standards for nondestructive examination
  - Thermal shield, if applicable.
- Containment vessel closures
  - Lid materials, and their dimensions and tolerances
  - Bolt specifications, including number, size, minimum thread engagement, and torque
  - Seal material, size, and compression specifications
  - Seal groove dimensions
  - Vent, drain, and leak-test ports, including closure methods.
- Impact limiters
  - Materials of construction and dimensions
  - Foam or wood specifications, including density
  - Method of attachment.

### A2.2.4 Typical Areas of Safety Review

- The general information review identifies the allowable contents, including water and other materials that could produce combustible gases.
- The structural and thermal reviews evaluate the performance of the containment system during both normal conditions of transport and hypothetical accident conditions. Primary emphasis is on the structural and thermal effects at the closure regions (lid and ports), including O-rings, plugs, and bolts, under hypothetical accident conditions.
- The structural and thermal reviews also verify the effects of the hypothetical accident conditions tests on the lead shielding and thermal shield (if applicable).
- The thermal review confirms the maximum temperature and pressure in the containment vessel under normal conditions of transport and hypothetical accident conditions.
- The containment review verifies that the package closures (lid, vent port, drain port) meet 10 CFR Part 71 containment criteria using the methods in ANSI N14.5 for both normal conditions of transport and hypothetical accident conditions. A typical maximum allowable

leakage rate is approximately $10^{-5}$ ref cc/s. The review also confirms that combustible-gas generation meets the criteria discussed in Section 4 of this document.

- The shielding review confirms that the package meets the allowable radiation levels during both normal conditions of transport and hypothetical accident conditions. The review should also confirm that the lead shielding does not melt under the hypothetical accident conditions.
- The criticality review verifies that the package contains either no fissile material, an exempt quantity of fissile material, or a fissile material quantity allowed under the general license provisions of 10 CFR Part 71. For packages with fissile content limited to quantities authorized by general license, the review also should confirm that the correct criticality transport index is specified.
- The review of operating procedures verifies that the bolts are properly torqued and that all penetrations of the containment vessel are properly leak tested prior to shipment. The review also addresses procedures that assure the contents are dry.
- The review of acceptance tests and maintenance program confirms that the appropriate leakage tests are performed for fabrication and periodic verification during the service life of the package. The review also ensures that appropriate acceptance testing of the lead shield and thermal performance is described and that the thermal performance of the packaging is maintained during the service life.

## References

American National Standards Institute, ANSI N14.5-1997, "American National Standard for Radioactive Materials–Leakage Tests on Packages for Shipment," New York.

# APPENDIX A3:

# UNIRRADIATED FUEL PACKAGES

## A3.1 Package Type

### A3.1.1 Purpose of Package

The purpose of this type of package is to transport unirradiated fuel assemblies and individual fuel rods. These packages are also referred to as "fresh fuel packages."

This appendix addresses only those packages in which the contents are limited to a Type A quantity of fissile material. For entire assemblies, this is typically achieved by restricting the enrichment. For individual fuel rods, a combination of enrichment and mass limits may be specified.

### A3.1.2 Description of a Typical Package

A typical packaging consists of a metal outer shell, closed with bolts and a weather-tight gasket. An internal steel strongback, shock-mounted to the outer shell, supports one or two fuel assemblies, which are fixed in position on the strongback by clamps, separator blocks, and end support plates. Depending on the type of fuel, neutron poisons are sometimes used to reduce reactivity. If the package is used to transport individual fuel rods, a separate inner container is often employed.

The contents of the package are unirradiated uranium in fuel assemblies or individual fuel rods. Because the majority of these packages are for commercial reactor fuel, the uranium is typically in the form of Zircaloy-clad uranium dioxide pellets.

Sketches of the typical package described above are presented in Figures A3-1 and A3-2.

### A3.1.3 Alternative Package Design

An alternative design for a fresh fuel package is shown in Figure A3-3. In this design, the fuel assemblies are fixed in position by two steel channels, mounted by angle irons or a similar bracing structure to a thin-walled inner metal container. This inner container is in turn surrounded by a honeycomb material and enclosed in a wooden outer container. Foam cushioning material is also generally used to cushion the fuel assemblies and may be used between the inner and outer container.

## A3.2 Package Safety

### A3.2.1 Safety Functions

The principal function of the package is to provide criticality control. The metal outer shell of the packaging retains the assemblies within a fixed geometry relative to other such packages in an array and provides impact and thermal protection. Shielding requirements are not significant because of the low radioactivity of unirradiated fuel.

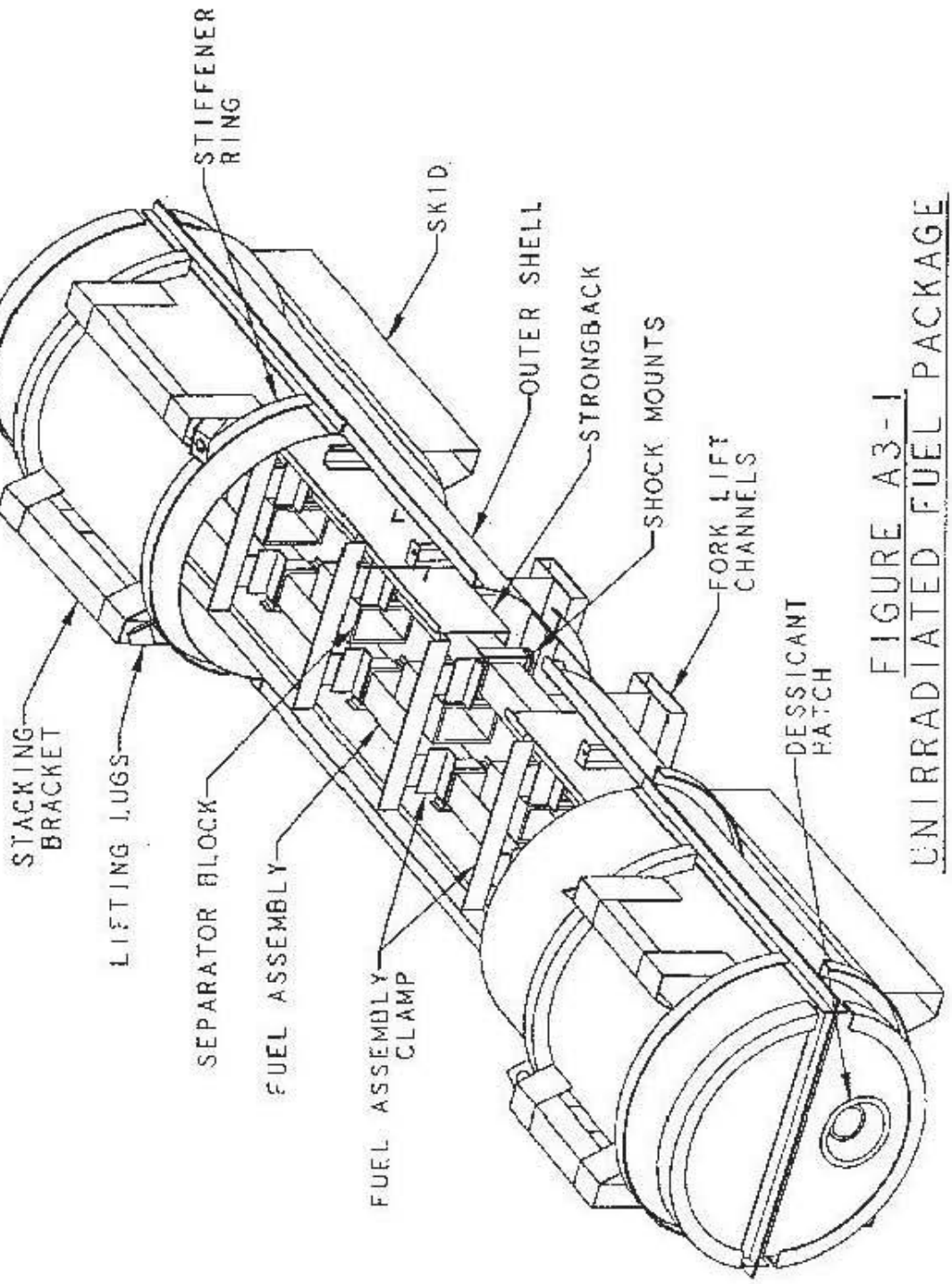


FIGURE A3-1
UNIRRADIATED FUEL PACKAGE

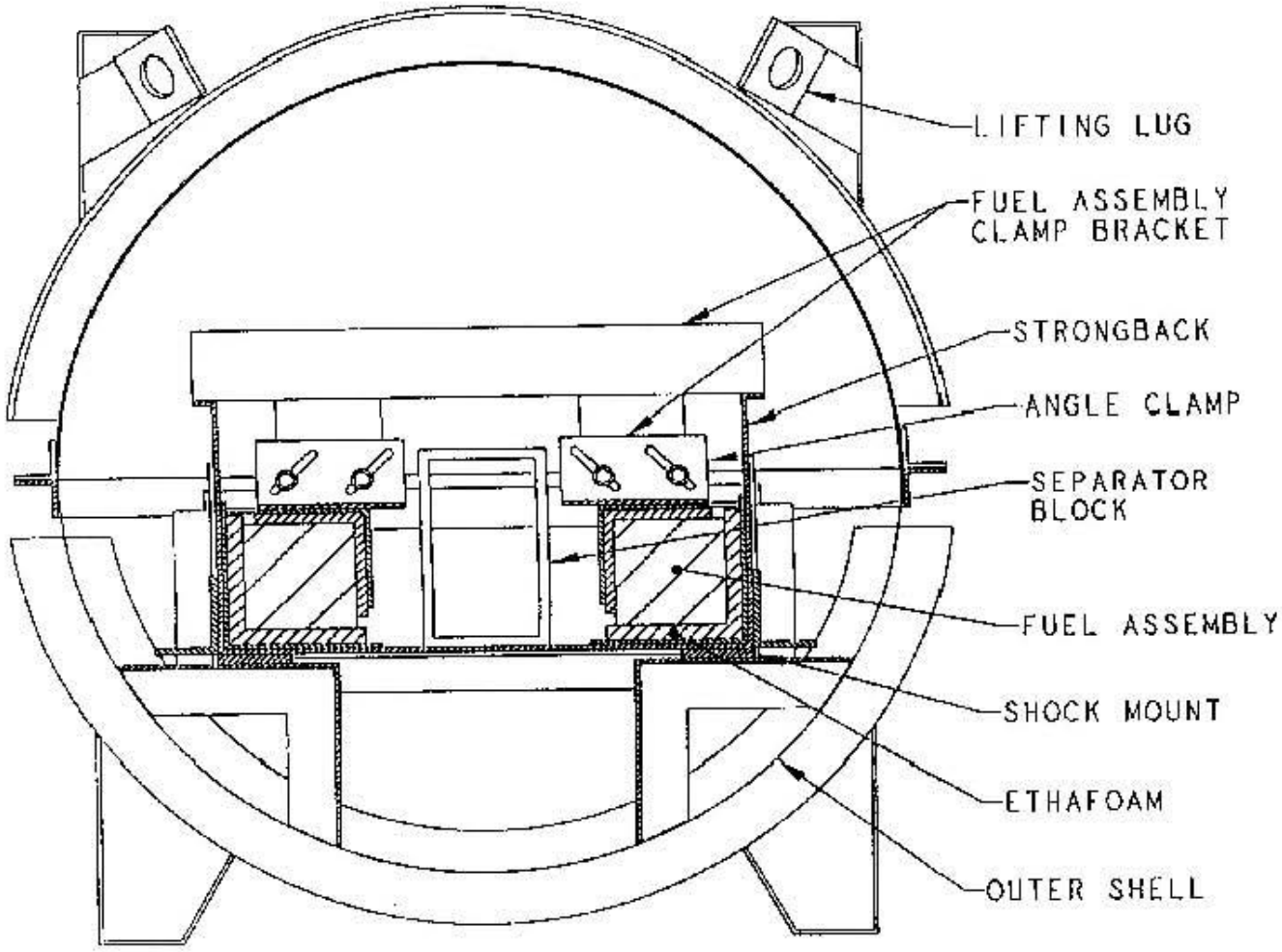


FIGURE A3-2
UNIRRADIATED FUEL PACKAGE
CROSS-SECTION WITH FUEL ASSEMBLIES

### A3.2.2 Safety Features

- A strongback with end support plates, clamps, and separators maintains the fuel assemblies in a fixed position relative to each other and to any neutron poisons.
- The metal outer shell of the packaging retains and protects the fuel assemblies, and may provide a minimum spacing between assemblies in an array of packages.
- Neutron poisons, if present, reduce reactivity.

### A3.2.3 Typical Areas of Review for Package Drawings

- Outer shell dimensions
- Structural components (e.g., strongback, support plates, fuel clamps and separators) that fix the position of fuel assemblies or relative position between fuel assemblies and poisons
  - Dimensions and materials
  - Methods of attachment.
- Neutron poisons
  - Dimensions and tolerances
  - Minimum poison content
  - Location and method of attachment.
- Moderating materials, including plastics, wood, and foam
  - Location
  - Material properties.

### A3.2.4 Typical Areas of Safety Review

- The general information review identifies the fuel assembly designs authorized in the package, including:
  - Number of and arrangement of fuel assemblies
  - Number, pitch, and position of fuel rods, guide tubes, and channels
  - Overall assembly dimensions, including active fuel length
  - Authorization or restrictions on missing fuel rods or partial-length rods
  - Maximum enrichment
  - Pellet dimensions and tolerances
  - Minimum cladding thickness
  - Fuel-clad gap

    - Type, location, and concentration of burnable poisons
    - Type, location, and quantity of plastics, such as polyethylene, within the fuel assemblies.
- The structural review addresses possible damage to the outer shell, strongback, fuel assembly, neutron poisons (if present), clamps, separators, and end support plates to ensure that the fuel assemblies and neutron poisons are maintained in a fixed position relative to each other under hypothetical accident conditions.
- The structural review also confirms the minimum spacing between fuel assemblies in different packages in an array under hypothetical accident conditions. Spacing can be affected by separation of the strongback from its shock mounts, failure of the shock mounts or fuel assembly clamps, and deformation of the outer shell of the package.
- The thermal review evaluates the effect of the fire on neutron poisons, plastic sheeting, wood, or other temperature-sensitive materials under hypothetical accident conditions.
- The criticality review addresses both normal conditions of transport and hypothetical accident conditions. Key areas for this review include:
    - The number of packages in the array and the array configuration (pitch, orientation of packages, etc.). Because of movement of the strongback within the package and the location of poisons, the arrays might not be symmetrical.
    - Degree of moderation. Structural features, as well as packaging material such as plastic sheeting, are evaluated for the possibility of differential flooding within the package. Plastic sheeting on the fuel assemblies should be open at both ends to preclude differential flooding. Flooding between the fuel pellets and cladding is also considered. Variations in the allowable amount of light-weight packaging material and plastic shims inserted in the fuel assemblies can also affect criticality under normal conditions of transport.
- The review of operating procedures ensures that instructions are provided so that proper clamps, separators, and poisons are selected for the type of fuel assemblies to be shipped and that these items are properly installed prior to shipment. The procedures should also address any other restrictions (e.g., limits on number of shims) considered in the package evaluation.
- The review of the acceptance tests and maintenance program verifies that the neutron poisons, if present, are subject to appropriate tests to verify their concentration and uniformity.

# APPENDIX A4:

# LOW ENRICHED URANIUM OXIDE PACKAGES

## A4.1 Package Type

### A4.1.1 Purpose of Package

The purpose of this type of package is to transport pellets and powder of low enriched uranium oxide. These packages are also referred to as "low enriched pellet and powder packages" or "oxide packages."

This appendix addresses only those packages in which the contents are limited to a Type A quantity of fissile material. This is achieved by limiting either the maximum enrichment or a combination of enrichment and mass.

### A4.1.2 Description of a Typical Package

A typical packaging consists of an inner steel vessel positioned within an outer steel drum. The outer drum, is typically a 30- or 55-gal. steel drum with a removable head and weather-tight gasket. The head is usually secured by a clamp ring with a closure bolt and a tamperproof seal. Vent holes near the top of the drum, which provide pressure relief under hypothetical accident conditions, are capped or taped during normal conditions of transport to prevent water inleakage.

The inner vessel is typically flanged, with a gasket and a bolted lid. The inner vessel is the containment vessel. It is centered in position inside the outer drum by foam, fiberboard, or similar insulation material. The inner vessel is not a pressure vessel and is not designed to prevent water inleakage under hypothetical accident conditions.

The contents of this package include low enriched uranium pellets, powder, and sometimes scrap, which are placed in plastic bags, metal cans, or cardboard boxes prior to loading into the inner container. Pellets are generally arranged on metal trays. Packages may include plates or liners with neutron poisons within the inner vessel. Spacers may be used within the inner vessel to maintain the position of the contents and to displace moderator in the event of water inleakage.

A sketch of a typical package for pellets or powder of low enriched uranium oxide is presented in Figure A4-1.

## A4.2 Package Safety

### A4.2.1 Safety Functions

The principal function of the package is to provide criticality control. The inner vessel provides containment to satisfy the requirements for Type A packages. Shielding requirements are not significant because of the low radioactivity of unirradiated uranium oxide.

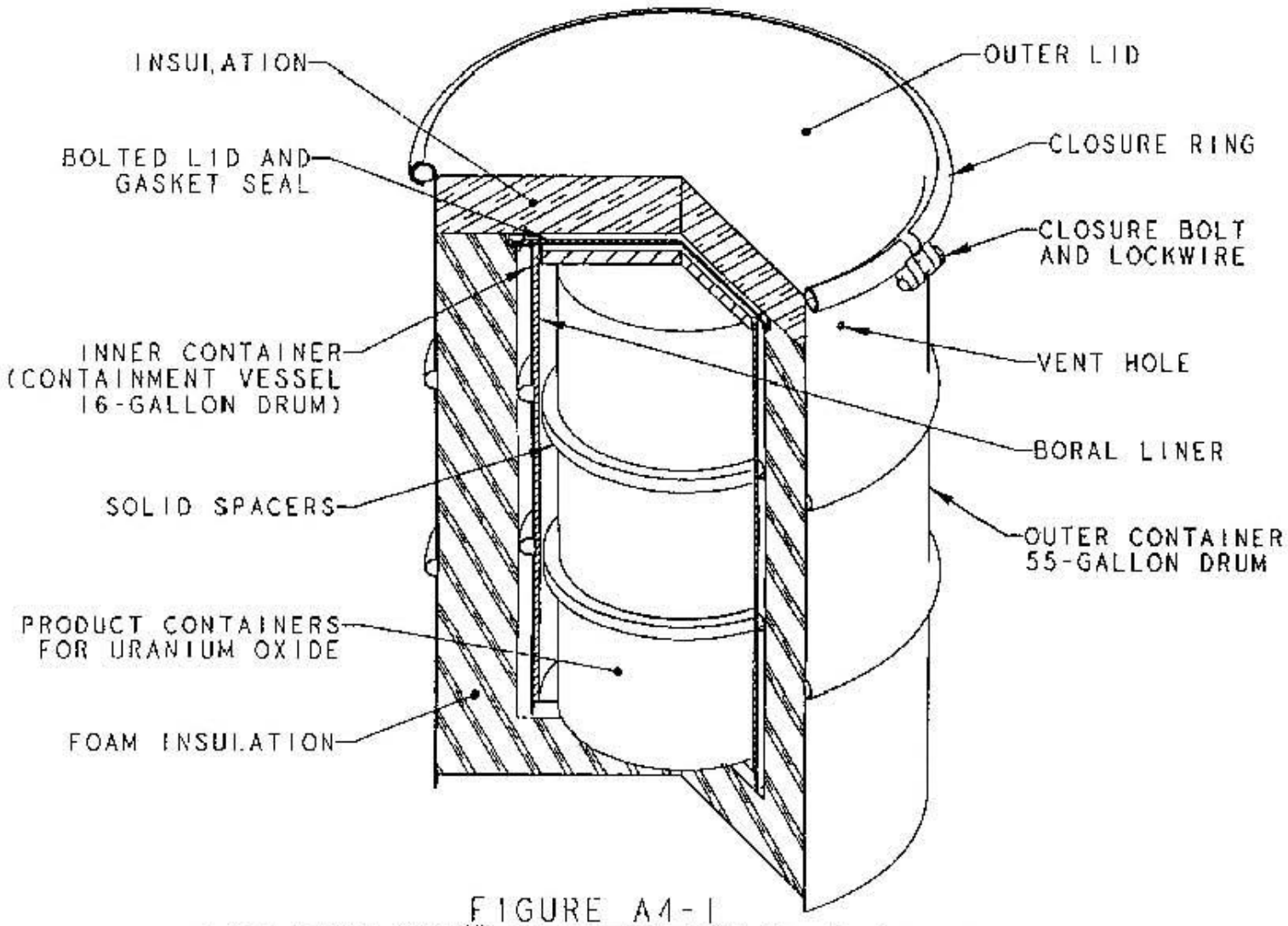


FIGURE A4-1
LOW ENRICHED URANIUM OXIDE PACKAGE

### A4.2.2 Safety Features

- The outer metal drum and insulation protect the inner vessel under hypothetical accident conditions, and maintain a minimum spacing between the inner containers of different packagings.
- The inner vessel provides containment and maintains a fixed geometry for criticality control.
- Neutron poisons, if present, reduce reactivity.

### A4.2.3 Typical Areas of Review for Package Drawings

- Inner vessel
  - Materials of construction
  - Dimensions and tolerances, including thickness
  - Product containers
  - Spacers, including materials and dimensions
  - Fabrication codes or standards.
- Neutron poisons
  - Isotopes and minimum concentration
  - Dimensions and tolerances
  - Location.
- Insulating material
  - Type
  - Dimensions and tolerances
  - Density.
- Outer drum
  - Material
  - Closure, including use of heavy-duty clamp ring, bolt torque
  - Dimensions.

### A4.2.4 Typical Areas of Safety Review

- The structural review evaluates package integrity under drop, puncture and thermal tests. This includes verifying that the lid of the outer drum remains in place and that the inner vessel is not damaged.

- The structural and thermal reviews address the minimum spacing between contents of different packages under hypothetical accident conditions. Damage to outer drum and charring of the insulation may result in closer spacing and more reactivity than that under normal conditions of transport.
- The thermal review also evaluates the effect of fire on neutron poisons and spacers.
- The criticality review addresses in detail both normal conditions of transport and hypothetical accident conditions. Key areas for this review include:
    - The configuration of the contents under normal conditions of transport and hypothetical accident conditions. This includes number, spacing, size, and condition of pellets, distribution of powders, and similar effects. Small changes in dimensions of the inner vessel can result in a significant increase in reactivity.
    - Distribution and degree of moderation. In addition to the moisture content of the pellets or powder, structural features, spacers, and packaging material such as plastic bags or cans are evaluated for the possibility of differential flooding within the package. Variations in the allowable amount of light-weight packaging material are also verified. Loading less than the maximum allowed contents can provide additional volume for water inleakage under hypothetical accident conditions, and therefore partial loads are often more reactive than a fully packed inner vessel.
    - The number of packages considered in the array and the array configuration (e.g., pitch and orientation of packages). Depending on the positioning of contents and the location of poisons, the arrays might not be symmetrical.
    - The degree and location of damage (e.g., drying or charring) to the thermal insulation caused by the fire test.
- The review of operating procedures ensures that instructions are provided so that proper neutron poisons or spacers are selected for the type of contents to be shipped and that the package is properly closed.
- The review of the acceptance tests and maintenance program verifies that the neutron poisons, if present, are subject to appropriate tests to verify their concentration.

# APPENDIX A5:

# TRANSURANIC WASTE PACKAGES

## A5.1 Package Type

### A5.1.1 Purpose of Package

The purpose of this type of package is to transport a Type B quantity of contact-handled transuranic waste.

### A5.1.2 Description of a Typical Package

A typical packaging consists of a stainless steel inner containment vessel housed inside a stainless steel and polyurethane outer containment assembly.

The outer containment vessel is a right circular cylinder with a flat bottom and domed lid. Its body and dome generally consist of polyurethane foam sandwiched between an inner and outer stainless steel shell. The dome-shaped lid is secured to the body by a locking ring. An elastomeric O-ring is used as the containment seal; a second O-ring allows the seal to be leak-tested. The assembly typically contains a leak-test port and a vent port. Fork pockets are often located at the base of the assembly for lifting and handling the entire package. Separate lifting devices are used for handling the lid only.

The inner containment vessel is a stainless steel shell with domed ends. The closure system consists of two O-rings, a leak-test port, and a vent port, similar to the outer containment vessel. Lifting devices on the inner lid can be used for lifting either the lid itself or an empty inner containment vessel.

The contents of the package consist of contact-handled transuranic waste produced primarily from plutonium production operations. The waste may be packaged within secondary containers. The contents may be limited to restrict the generation of hydrogen or other combustible gases.

Several packages may be secured to a special trailer for transport.

A sketch of a typical transuranic waste package is presented in Figure A5-1.

## A5.2 Package Safety

### A5.2.1 Safety Functions

The principal safety functions of the package are to provide containment and criticality control.

### A5.2.2 Safety Features

- The inner and outer containment vessels provide double containment for the plutonium.

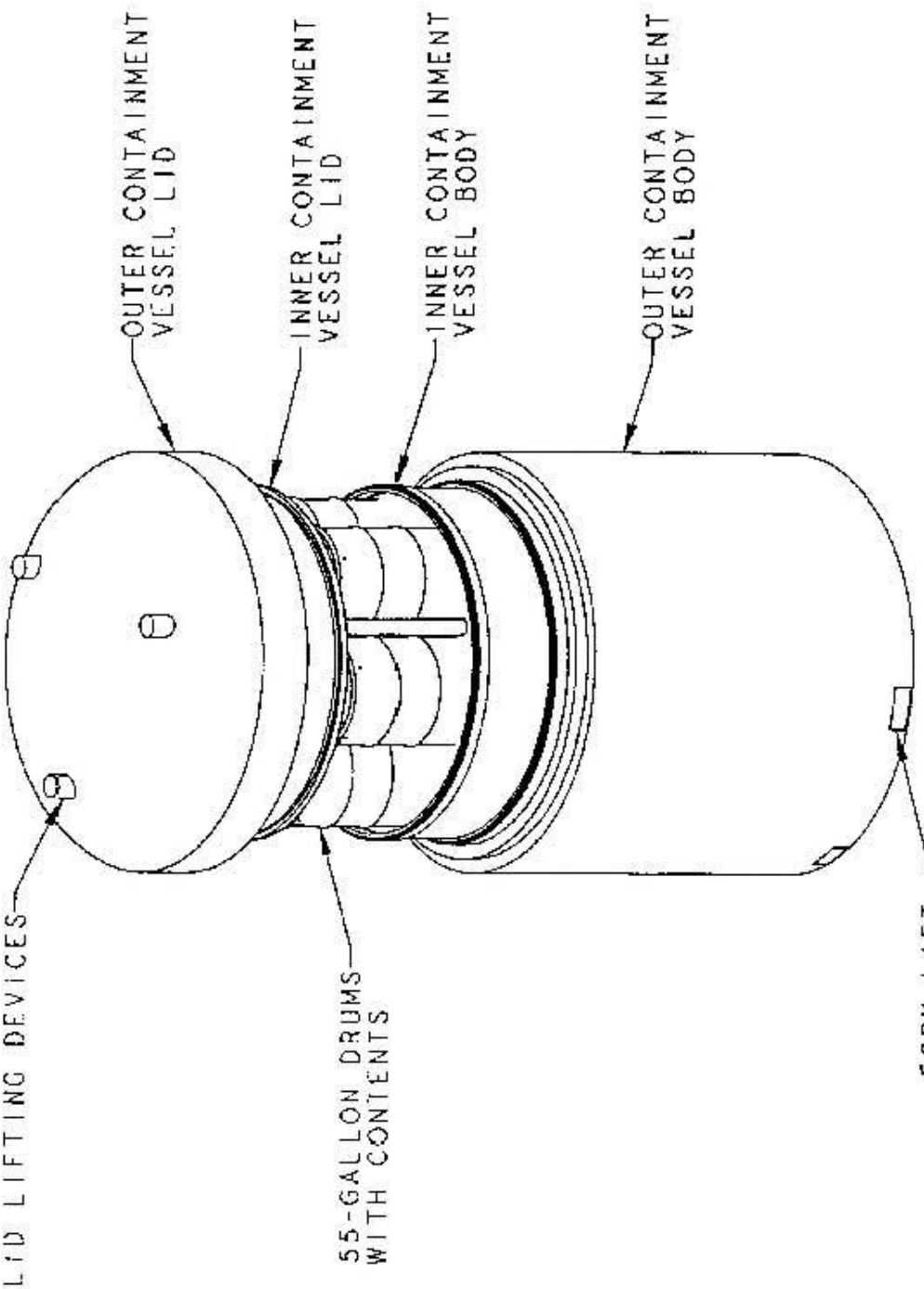


FIGURE A5-1
TRANSURANIC WASTE PACKAGE

- The steel package and configuration of the secondary containers provide sufficient attenuation and distance from the waste to satisfy the shielding requirements for normal conditions of transport (exclusive use) and hypothetical accident conditions.
- The limit on the allowed mass of fissile material provides criticality control for a single package. The physical size and separation of contents also ensures subcriticality for arrays.

### A5.2.3 Typical Areas of Review of Package Drawings

- Containment vessels
  - Materials of construction
  - Dimensions and tolerances
  - Fabrication codes or standards
  - Weld specifications, including codes or standards for nondestructive examination
  - Foam specification and density, as applicable.
- Containment vessel closures
  - Lid materials, and their dimensions and tolerances
  - Closure device design details, such as bolt specifications and torque
  - Seal material, size, and compression specifications
  - Seal groove dimensions
  - Vent and leak-test ports, including closure methods.

### A5.2.4 Typical Areas of Safety Review

- The structural and thermal reviews evaluate the ability of the containment vessels to perform their intended functions under normal conditions of transport and hypothetical accident conditions. Primary emphasis is on the structural effects near the O-ring regions (including closure devices) and on the thermal performance of the O-rings.
- The thermal and containment reviews verify that the hydrogen concentration in any confined volume will not exceed 5% (by volume) during a period of one year. Shorter time periods have been approved based on detailed operating procedures to control and track the shipment of packages. The reviews also should ensure that the containment evaluation specifies that the secondary containers are aspirated prior to shipment.
- The containment review verifies that the 10 CFR Part 71 containment criteria are satisfied for both normal conditions of transport and hypothetical accident conditions. With typical contents, the package must remain leaktight, as defined in ANSI N14.5. Each containment vessel must separately meet the 10 CFR Part 71 containment criteria.

- The shielding review evaluates the ability of the package to satisfy the allowed radiation levels during normal conditions of transport and hypothetical accident conditions.
- The criticality review confirms that a single package and array of packages are subcritical during both normal conditions of transport and hypothetical accident conditions.
- The review of operating procedures verifies that any free-standing water is removed from both containment vessels and that they are closed and leak-tested prior to shipment. The review also typically ensures that the secondary containers are aspirated prior to shipment.
- The review of acceptance tests and maintenance program verifies that appropriate fabrication and periodic verification leakage tests are performed.

## References

American National Standards Institute, ANSI N14.5-1997, "American National Standard for Radioactive Materials–Leakage Tests on Packages for Shipment," New York.

# APPENDIX A6:

# LOW ENRICHED URANIUM HEXAFLUORIDE PACKAGES

## A6.1 Package Type

### A6.1.1 Purpose of Package

The purpose of this type of package is to transport low-enriched solid uranium hexafluoride ($UF_6$).

### A6.1.2 Description of a Typical Package

A typical packaging consists of an inner steel cylinder that acts as a containment vessel, and an outer protective overpack. Unenriched $UF_6$ may be transported in bare cylinders, without the protective overpack, as authorized in DOT regulations. Protective overpacks are typically required only for the transport of enriched (fissile) $UF_6$. ANSI N14.1 specifies the design and fabrication of the $UF_6$ cylinder. ANSI N14.1 and USEC-651 contain information regarding overpacks.

The inner cylinder is carbon steel, with rounded ends and a protective skirt. On one end of the cylinder is a valve for filling and emptying the cylinder; on the other end is a removable plug. The most commonly used commercial cylinders are approximately 0.76 m (30 in.) in diameter, 2.1 m (81 in.) in length, with a capacity of about 2300 kg (2.5 tons) of $UF_6$. The design and authorized contents are defined in ANSI N14.1.

The protective overpack is generally a double-shell, stainless steel cylinder with cushioning pads on the inner cavity. An energy-absorbing, insulating foam fills the space between the inner and outer shell. The overpack can be separated into two halves to enable easy access to the inner cylinder. Overpacks for the 30-in. cylinders mentioned above are approximately 0.016 m (4 in.) thick.

For the 30-in. cylinder, the $UF_6$ enrichment must not exceed 5%. The cylinder is filled with liquid $UF_6$. Because of the volume reduction during cooling and solidification of the $UF_6$, the final internal pressure is less than one atmosphere in the cylinder.

A sketch of a typical $UF_6$ package (cylinder and overpack) is presented in Figure A6-1.

## A6.2 Package Safety

### A6.2.1 Safety Functions

The primary function of the package is to provide containment and moderation control for criticality purposes. Moderation control is required for all commercially used cylinders for fissile $UF_6$ and must be maintained under normal conditions of transport and hypothetical accident conditions. To assure subcriticality by moderation control, the mass of the contents must be at least 99.5% $UF_6$.

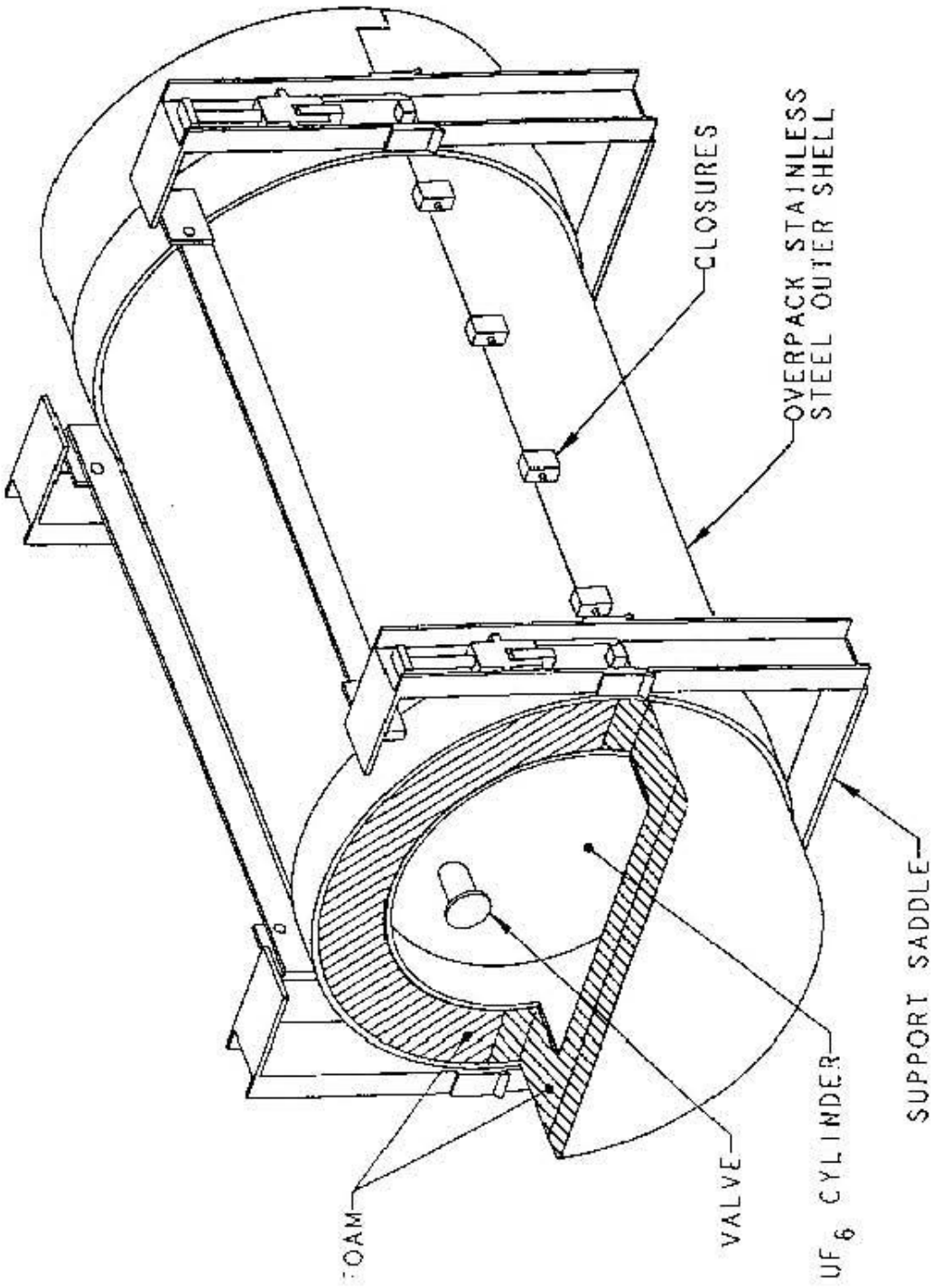


FIGURE A6-1
URANIUM HEXAFLUORIDE PACKAGE

The cylinder is defined as the containment boundary for the $UF_6$. Unirradiated uranium enriched to less than 5% is a Type A quantity. Recycled uranium can be a Type B quantity due the presence of $^{232}U$, $^{234}U$, $^{236}U$, and various radioactive impurities.

Shielding requirements are generally not significant because of the low radioactivity and self-shielding of $UF_6$. Compliance with regulatory limits for radiation levels is verified prior to shipment.

The overpack provides thermal protection to prevent overheating of the $UF_6$, which can cause hydraulic failure of the cylinder. The overpack also provides impact protection for the cylinder and the valve.

### A6.2.2 Safety Features

- The steel cylinder precludes inleakage of water and provides containment under normal conditions of transport and hypothetical accident conditions.
- The cylinder skirt provides some protection to the valve during handling operations, normal conditions of transport, and hypothetical accident conditions.
- The overpack provides structural and thermal protection for the cylinder and its valve under hypothetical accident conditions.

### A6.2.3 Typical Areas of Review for Package (Overpack) Drawings

- Overpack shell
  - Materials of construction
  - Dimensions and tolerances
  - Vents for pressure-relief of foam combustion products.
- Foam specifications
  - Type
  - Density
  - Compressive strength
  - Fire retardant characteristics
  - Limit on free chlorides.
- Closure devices
  - Torque
  - Valve protection device.

## A6.2.4 Typical Areas of Safety Review

- The structural review concentrates on the ability of the overpack to protect the valve under hypothetical accident conditions.
- The structural and thermal reviews address the ability of the overpack to provide protection to the cylinder itself under hypothetical accident conditions. Because of the heat capacity of the $UF_6$, a partially filled cylinder may be more susceptible to hydraulic failure than a full cylinder.
- The containment review verifies that the cylinder meets the containment criteria in ANSI N14.5 for Type B packages.
- The criticality review confirms that there is no water inleakage under normal conditions of transport and hypothetical accident conditions. The minimum transport index for criticality control is specified in 49 CFR 173.417.
- The review of operating procedures ensures that the valve is properly closed and leak tested, as appropriate, and that the valve protection device, if applicable, is installed. This review also confirms that the radiation levels are verified to meet the regulatory limits prior to transport.
- The review of the acceptance tests and maintenance program evaluates the inspection procedures for the overpack, including the physical condition of the inner and outer shells, corrosion, performance of the foam during the service life of the overpack, and wear of cushioning pads between the cylinder and overpack. The review also verifies that the cylinder is tested and maintained in accordance with the requirements in 49 CFR 173.420 and ANSI N14.1.

## References

American National Standards Institute, ANSI N14.5-1997, "American National Standard for Radioactive Materials–Leakage Tests on Packages for Shipment," New York.

Institute of Nuclear Materials Management, "American National Standard for Nuclear Materials—Uranium Hexafluoride–Packaging for Transport," ANSI N14.1-1995, New York.

U. S. Enrichment Corporation, "Uranium Hexafluoride: A Manual of Good Handling Practices," USEC-651 (Revision 7), January 1995.

# APPENDIX A7:

# HIGH ENRICHED URANIUM OR PLUTONIUM PACKAGES

## A7.1 Package Type

### A7.1.1 Purpose of Package

The purpose of this type of package is to transport Type B quantities of high enriched uranium or plutonium (other than by air).

### A7.1.2 Description of a Typical Package

A typical packaging consists of one or two containment vessels and an outer container. Double containment is required for plutonium in excess of 20 Ci, except as specified in §§71.63(b)(1-3).

The outer container is a steel drum with a removable head and weather-tight gasket. The head is usually secured by a clamp ring with a tamperproof seal. Vent holes near the top of the drum, which provide pressure relief under hypothetical accident conditions, are capped or taped during normal conditions of transport to prevent water inleakage.

The inner containment vessel is a steel container, typically a stainless steel cylinder, with a maximum outer diameter of 0.127 m (5 in.), closed by a welded bottom cap and a welded top flange with a bolted lid. The lid, which is sealed by two O-rings, contains a leak-test port and sometimes a separate fill port for leak testing. Unless double containment is provided, this containment vessel is centered in position inside the outer container by fiberboard (or similar material) insulating material. If double containment is required, the inner (primary) containment vessel is positioned inside a secondary containment vessel.

The contents are uranium or plutonium, typically in metal, oxide, or nitrate form. The uranium or plutonium is generally placed in plastic bags or metal cans prior to loading into the containment vessel. Spacers are often used to maintain the position of the contents. Uranium may be in liquid form. Plutonium in excess of 20 Ci must be shipped as a solid.

A sketch of a typical package for high enriched uranium is presented in Figure A7-1. A package for plutonium would be similar, except than a second containment system would be required.

## A7.2 Package Safety

### A7.2.1 Safety Functions

The principal functions of the package are to provide containment and criticality control.

Package design features that accomplish the containment and criticality functions generally also provide adequate shielding to satisfy the requirements for nonexclusive-use shipment. Additional shielding may

be required if significant quantities of certain isotopes, e.g., $^{238}Pu$ or $^{241}Am$ (from the decay of $^{241}Pu$) are present.

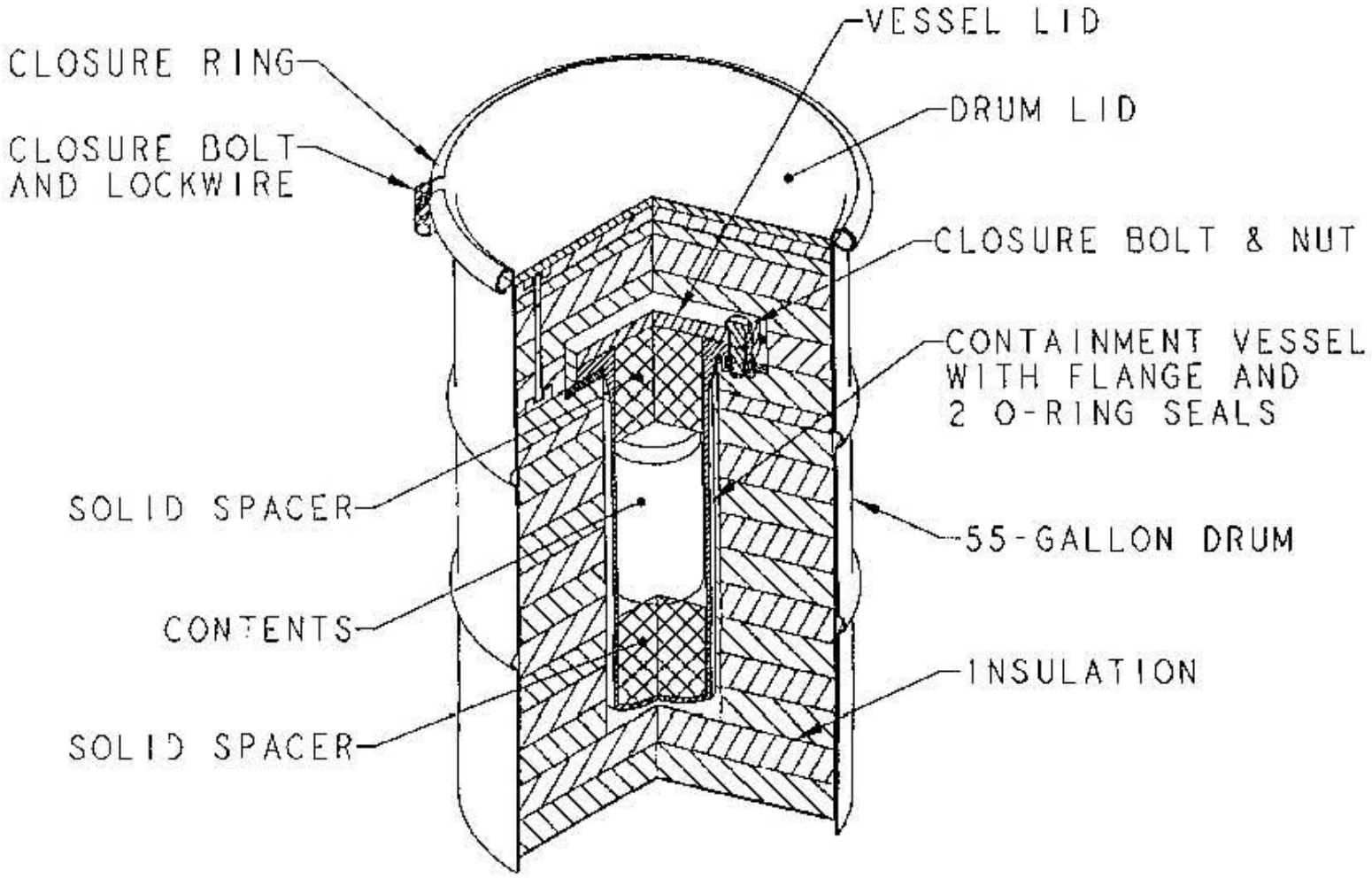


FIGURE A7-1
HIGH ENRICHED URANIUM PACKAGE

### A7.2.2 Safety Features

- The steel drum and insulating material protect the containment vessel and contents under hypothetical accident conditions and maintain a minimum spacing between packagings for criticality control.
- The steel inner vessel provides containment of the radioactive material. An additional containment vessel also provides containment for plutonium, if required by §71.63(b).
- The diameter and volume of the inner containment vessel, together with limits on the fissile mass of the contents, ensure that a single package is subcritical.
- The containment vessel, insulating material, and steel drum maintain a minimum distance from the contents to the package surface and provide some attenuation to satisfy the shielding requirements.

### A7.2.3 Typical Areas of Review for Package Drawings

- Containment vessel body
  - Materials of construction
  - Dimensions and tolerances, including maximum cavity dimensions
  - Fabrication codes or standards
  - Weld specifications, including codes or standards for nondestructive examination.
- Containment vessel closures
  - Lid materials, dimensions, and tolerances
  - Bolt specifications, including number, size, and torque
  - Seal material, size, and compression specifications
  - Seal groove dimensions
  - Leak-test ports.
- Spacers to position or displace fissile material
  - Material of construction
  - Dimensions and tolerances
  - Locations.
- Insulating material
  - Type
  - Dimensions and tolerances
  - Density.

- Outer drum
  - Material
  - Closure, including use of heavy-duty clamp ring, bolt torque
  - Dimensions
  - Applicable codes or standards.

### A7.2.4 Typical Areas of Safety Review

- The structural review confirms that packaging integrity is maintained under the drop, crush, and puncture tests. The review also verifies that the drum lid remains securely in place.
- The structural and thermal reviews evaluate the performance of the containment system under both normal conditions of transport and hypothetical accident conditions. Primary emphasis is on the structural integrity of the inner vessel and its closure, and on the thermal performance of the O-rings. If the package provides double containment, each containment vessel must separately meet the containment criteria.
- The structural and thermal reviews address the condition of the package and the minimum spacing between different packages under hypothetical accident conditions. Damage to the outer drum and charring of the insulating material may result in closer spacing than that of normal conditions of transport.
- The thermal and containment reviews verify that the hydrogen concentration in any confined volume will not exceed 5% (by volume) during a period of one year. Shorter time periods have been approved based on detailed operating procedures to control and track the shipment of packages.
- The criticality review addresses in detail both normal conditions of transport and hypothetical accident conditions. Key parameters for this review include the number of packages in the arrays, array configuration (pitch, orientation of packages, etc.), positioning of the containment vessels within the drum, moderation due to inleakage of water, the condition and quantity of spacing material, and interspersed moderation between packages.
- The review of operating procedures confirms that the containment vessels have been properly closed and bolts torqued, and that an appropriate pre-shipment leak test is performed.
- The review of the acceptance tests and maintenance program verifies that appropriate fabrication and periodic verification leakage tests are performed.

# APPENDIX A8:

# TYPE B SPECIAL FORM PACKAGES

## A8.1 Package Type

### A8.1.1 Purpose of Package

The purpose of this type of package is to transport a Type B quantity of radioactive material in special form.

### A8.1.2 Description of a Typical Package

A typical packaging consists of a cask body with a lid, base, and protective jacket.

The cask body is a lead-filled cylinder with a stainless steel inner and outer shell. A drain tube penetrates the cavity and is sealed with a plug, which is covered by the protective jacket during transport. A lead-filled, stainless steel lid is bolted to the tapered top of the main body and sealed with a weather-tight gasket. Both the body and the lid generally have lifting devices that are covered during shipment by the protective jacket.

The base is a square steel skid that bolts to the protective jacket. The skid consists of energy-absorbing steel angles (stiffeners). Several I-beams are welded to the base to enable handling by a forklift.

The protective jacket is a double-walled steel cylinder with an open bottom and a protruding box section positioned diametrically across the top and vertically down the sides. The jacket may contain thermal insulation. A steel flange bolts to the base, and the main body of the packaging is centered within the jacket by steel tubes welded to the jacket inner wall. Steel lifting loops are typically welded to the top corners, and tie-down devices are welded to the sides.

The contents of the package typically consists of byproduct material in special form.

A sketch of a typical Type B special form package is presented in Figure A8-1.

## A8.2 Package Safety

### A8.2.1 Safety Functions

The principal safety function of the package is to provide radiation shielding. Containment is provided primarily by the special form source itself. The packaging must maintain the sources in the fully shielded configuration under normal conditions of transport and hypothetical accident conditions.

### A8.2.2 Safety Features

- The lead shield provides shielding for gamma radiation.
- The protective jacket provides structural and thermal protection to the main body, which contains the special form radioactive material.

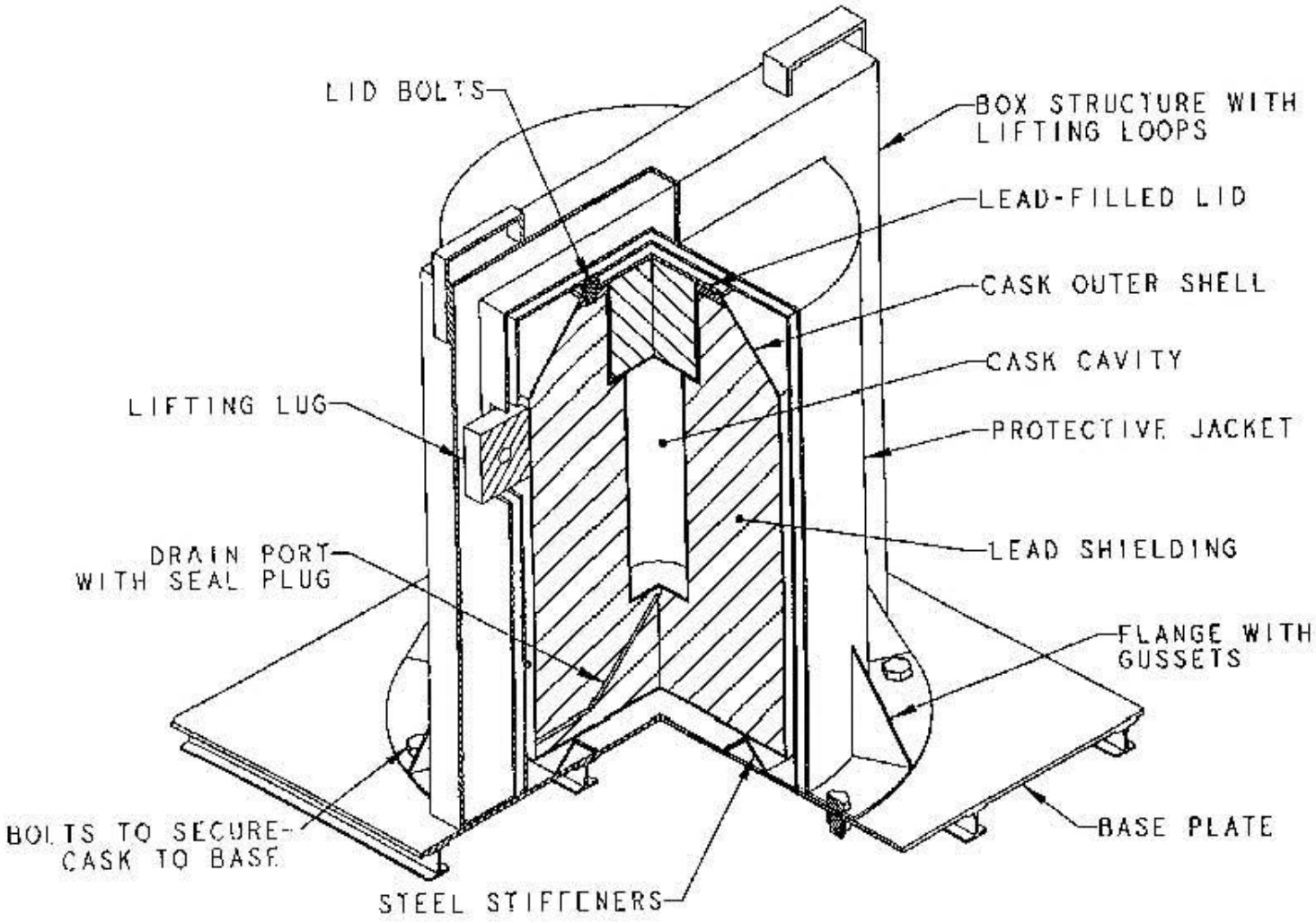


FIGURE A8-1
TYPE B SPECIAL FORM PACKAGE

### A8.2.3 Typical Areas of Review for Package Drawings

- Cask body
  - Materials of construction
  - Dimensions and tolerances of steel shells and lead shield
  - Fabrication codes or standards, including any special processes for lead pour
  - Weld specifications, including codes or standards for nondestructive examination.
- Closures
  - Lid materials, and their dimensions and tolerances
  - Bolt specifications, including number, size, minimum thread engagement, and torque
  - Seal material, size, and compression specifications
  - Seal groove dimensions
  - Vent and leak-test ports, including closure methods.
- Protective jacket
  - Method of attachment
  - Bolt specifications, including number, size, minimum thread engagement, and torque
  - Insulating material.

### A8.2.4 Typical Areas of Safety Review

- The review of the general information verifies that the contents are special form.
- The structural and thermal reviews evaluate the ability of the shield to perform its intended function under normal conditions of transport and hypothetical accident conditions. Lead slumping should be inconsequential and the lead should not melt. These reviews ensure that the package has been tested under the most damaging conditions (e.g., impact orientation). The integrity of the cask closure and bolts is also reviewed.
- The thermal review should verify that no credit has been taken for the presence of helium in gaps between packaging components. The review should verify that the heat transfer medium is air, and that the effects of air on the contents and packaging components have been addressed.
- The shielding review evaluates the ability of the package to satisfy the allowed radiation levels during both normal conditions of transport and hypothetical accident conditions.
- The review of operating procedures verifies that the cask has been appropriately drained and that the bolts are properly torqued.
- The review of the acceptance tests and maintenance program ensures that appropriate tests are specified for shielding and thermal performance.