Strål
säkerhets
myndigheten
Swedish Radiation Safety Authority

Author: Dennis Mennerdahl

Technical Note

2013:16

Assessment of PWR fuel depletion and of neutron multiplication factors for intact PWR fuel copper canisters

Main review phase

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SSM perspektiv

Bakgrund

Strålsäkerhetsmyndigheten (SSM) granskar Svensk Kärnbränslehantering AB:s (SKB) ansökningar enligt lagen (1984:3) om kärnteknisk verksamhet om uppförande, innehav och drift av ett slutförvar för använt kärnbränsle och av en inkapslingsanläggning. Som en del i granskningen ger SSM konsulter uppdrag för att inhämta information och göra expertbedömningar i avgränsade frågor. I SSM:s Technical note-serie rapporteras resultaten från dessa konsultuppdrag.

Projektets syfte

Det övergripande syftet med projektet är att ta fram synpunkter på SKB:s säkerhetsanalys SR-Site för den långsiktiga strålsäkerheten hos det planerade slutförvaret i Forsmark. Detta uppdrag avser granskning av nukleär kriticitetssäkerhet.

Författarens sammanfattning

SKB (Svensk Kärnbränslehantering AB) ansökte 2011 om svenska regeringens tillstånd för en föreslagen lösning för slutförvaring av använt bränsle från svenska kärnkraftverk och en del mindre kvantiteter av annat fissilt material.

Denna Technical Note innehåller resultat från en färsk genomgång av SKB:s metoder for att bestämma inverkan av reaktorutbränning på keff för intakt PWR-bränsle i en intakt kopparkapsel flödad med vatten. Granskningen har, så lång möjligt, baserats på allmän information i ansökan, till skillnad mot den utvalda (och möjligen bearbetade) information som finns I utvärderingen av kriticitetssäkerhet. Beräkningsmetoderna (datorprogram och tvärsnittsdata) vid granskningen har varit annorlunda än de som använts av SKB men det finns gemensamma felkällor. Tillämpbarheten av granskningsmetoden baseras primärt på benchmarks för keff och reaktivitet, till skillnad från benchmarks baserade på materialsammansättningar som använts av SKB.

Granskningen har fokuserats på reaktorutbränning av PWR-bränsle och av intakt kopparkapsel för intakt PWR-bränsle. Specifika händelseförlopp kan utvärderas senare, med denna information som grund. Nyligen (2011) publicerade benchmarks från Electric Power Research Institute (EPRI) har utnyttjats. Tidigare och pågående studier av utbränningskreditering av OECD/NEA ger värdefull information om utbränningsberäkningar från testade metoder. En pågående utvärdering av OECD/NEA (för IRPhE-handboken med benchmarks) av mätningarna som ligger till grund för EPRI.s benchmarks är värdefull (ett utkast förväntas bli publicerat vården 2013).

Det övergripande resultatet är att det inte tycks finnas något större hinder vid bestämning av keff noggrant både för färskt och för bestrålat PWR-bränsle i en intakt PWR-kapsel. Resultaten i SKB:s ansökan är trovärdiga, inom de osäkerheter som specificeras. Nyligen publicerade och pågående utvecklingsprojekt avseende benchmarks (både mätningar av integral reaktivitet och av materialsammansättningar) för utbränningsmetoder kan tillämpas för att minska osäkerheten ytterligare.

Projektinformation
Kontaktperson på SSM: Mikael Kjellberg
Diarienummer ramavtal: SSM2011-4444
Diarienummer avrop: SSM2012-3834
Aktivitetsnummer: 3030007-4038

SSM perspective

Background

The Swedish Radiation Safety Authority (SSM) reviews the Swedish Nuclear Fuel Company's (SKB) applications under the Act on Nuclear Activities (SFS 1984:3) for the construction and operation of a repository for spent nuclear fuel and for an encapsulation facility. As part of the review, SSM commissions consultants to carry out work in order to obtain information and provide expert opinion on specific issues. The results from the consultants' tasks are reported in SSM's Technical Note series.

Objectives of the project

The general objective of the project is to provide review comments on SKB's postclosure safety analysis, SR-Site, for the proposed repository at Forsmark.

Summary by the author

SKB (Svensk Kärnbränslehantering AB) in 2011 applied to the Swedish government for approval of a proposed solution for disposal of used fuel from Swedish nuclear power reactors and some relatively minor quantities of other fissile material.

This Technical Note contains results of a recent review of the SKB methods used to determine the influence of reactor depletion on keff of intact PWR fuel in an intact PWR copper canister flooded with water.

The review has, as far as possible, been based on general information in the licensing application, as opposed to the selected (and possibly modified) information in the criticality safety evaluation. The review calculation methods (computer codes and cross-section data) have been different than the methods used by SKB, but there are some common error sources.

The review method validation is primarily based on keff and reactivity benchmarks rather on composition benchmarks as used by SKB. The review has focused on the reactor depletion of PWR fuel and of the intact fuel in the intact PWR copper canister. Specific scenarios can be evaluated later, with this information as a basis. Advantage has been taken of recent (publication 2011) Electric Power Research Institute (EPRI) benchmarks. Past and on-going burnup credit studies by OECD/NEA provide valuable information on depletion calculations, often from validated methods. An on-going evaluation by OECD/NEA (for the IRPhE handbook with benchmarks) of the measurements involved in the EPRI benchmarks is valuable (a draft is expected to be published in the spring of 2013).

The overall result is that there appears to be no major obstacles in determining keff quite accurately both for fresh and for depleted PWR fuel in an intact PWR canister. The results in the SKB application are credible, within the uncertainties specified. Recently published and on-going validation developments for depletion method benchmarks (both measurements of integral reactivity and of material compositions) might be applied to reduce the uncertainty further.

Project information

Contact person at SSM: Mikael Kjellberg

Strål
säkerhets
myndigheten

Swedish Radiation Safety Authority

Author: Dennis Mennerdahl
E Mennerdahl Systems, Sweden

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Contents

1. Introduction **3**
2. Nuclear criticality safety criteria **4**
3. PWR fuel specification[1] **4**
3.1. PWR fuel types and specifications 4
3.2. Unirradiated PWR fuel assembly contents 8
3.3. Depleted PWR fuel assemblies 13
3.4. Selection of PWR fuel assemblies for canister 14
3.5. Burnup determination and assembly selection 15
3.6. Burnup loading curves 16
3.7. Canister types I, II and III 17
3.7.1. The PWR I type-canister 18
3.7.2. The PWR II type-canister 19
3.7.3. The PWR III type canister 19
3.7.4. The PWR-MOX canister 20
4. Reactor operating data for the fuel **20**
5. The copper canister specifications **23**
6. Additional specifications in the SKB criticality safety report **33**
7. SKB calculation methods **38**
8. SKB calculations for the PWR canister **40**
9. EMS calculation methods **42**
9.1. General description of methods 42
9.2. Validation standards for calculation methods 43
9.3. Validation of k_{eff} calculation methods 45
9.4. Validation of fuel depletion methods 45
9.5. Validation of SCALE 6.1.1 k_{eff} calculations 46
9.6. Validation of SCALE 6.1.1 depletion calculations 47
9.6.1. OECD/NEA calculation benchmarks 47
9.6.2. EPRI and IRPhEP benchmarks 48
9.6.3. Recent NRC SFST ISG-8 Rev.3 validation approach 48
9.6.4. Summary of EMS validation of burnup credit methods 48
10. EMS review calculations **49**
10.1.1. Purpose 49
10.1.2. Canister design 49
10.1.3. PWR fuel 51
10.1.4. PWR depletion in Ringhals 2, 3 and 4 52
10.1.5. PWR depletion models 52
10.1.6. PWR canister models 53
10.1.7. Calculation cases 54
10.1.8. Calculation results 55
11. Comparisons of EMS and SKB results **57**
12. The Consultant's assessment **58**
13. References **60**
Appendix A. OECD/NEA/NSC/WPNCS **62**
A.1. Phase I-B 62
A.2. Phase III-B 63
A.3. Phase IV-B 65
A.4. Phase II-D 67
A.5. Phase III-C 70
Appendix B. EPRI benchmarks **71**
B.1. Introduction 71
B.2. Case 1 72

Appendix C. IRPhE Handbook benchmark 75
Appendix D. Coverage of SKB reports 76

1. Introduction

The Swedish SKB (Svensk Kärnbränslehantering AB) has used burnup credit to support the nuclear criticality safety of the proposed final disposal of Swedish nuclear power reactor fuel. Burnup credit is a criticality safety term that refers to taking credit for a reduction in the multiplication factor (k_{eff}, an inverse eigenvalue in the neutron transport equation) due to depletion (transmutation) of the nuclear fuel in the reactor and later radioactive decay. Burnup credit is a criticality safety control implemented by management decision.

This Technical Note is not a nuclear criticality safety review. However, it covers important components of a nuclear criticality safety review of an intact water-filled copper canister for PWR fuel. Important issues include:

- The specifications of the fuel
- The specifications of the fuel history (primarily during reactor operation)
- The specifications of the canister
- Selection of a calculation method
- Validation of the calculation method
- Independent calculations of depletion effects and of k_{eff} values
- Comparison with the SKB safety assessment
- Further suggestions for criticality safety review

In addition to the burnup credit for the copper canister, SKB currently applies burnable absorber (BA) credit to BWR fuel in the CLAB facility. BA credit is not burnup credit. Depletion consideration is required for used BWR fuel with BA credit. Depletion consideration related to burnup credit for used BWR fuel with or without BA is optional.

There are many other systems than the intact canister with PWR fuel and there are other issues that need to be included in a complete criticality safety review. The other issues include impact of the human factor (e.g. misloading of fuel), impact of potential incidents, damage and long-term degradation to the fuel and to the canister. They need to be reviewed later since this initial phase focuses on calculation methods and accuracy in determining k_{eff} for the intact, water-filled PWR canister.

The results should be applicable to future reviews involving burnup credit and BA credit for used nuclear fuel.
Chapters 2-8 contain information, often directly copied, from the SKB source documents. All information in this report is based on publicly available information. That information has been screened to fit the purpose of the report and apparent editorial errors and inconsistencies have been corrected. Sometimes the text is an interpretation of the information in the source documents. There may thus be some differences between information in chapters 2-8 compared with the source documents.

The compilation of information from the safety documentation is a part of the review method. That compilation has not been reviewed in detail and should not be used as a source for other work or for safety-related decisions.

2. Nuclear criticality safety criteria[1]

The SKB criticality safety design criteria are summarized in chapter 3, section 2.1.2 of SKBdoc 1091554 (reference 1):

”Anläggningen, dess system och komponenter ska konstrueras att motstå felfunktion, yttre och inre belastningar så att en händelse som kan leda till en radiologisk olycka med radioaktivt utsläpp har en frekvens som är mindre än 10^{-6}/år. Detta innebär att för alla konstruktionsstyrande händelser ska kopparhöljets täthet bibehållas.”

The author’s translation of the criticality safety requirement in the same section 2.1.2 is: “The canister internal geometry and the contained fuel shall for all design basis events for the repository comply with a safety margin such that $k_{eff} < 0.95$.”

Appendix 1 (page 2) in SKBdoc 1091152 (reference 2): ”Kriticitet ska under inga förutsättningar kunna uppstå, oavsett hur bränslet är disponerat i kapseln (krav på inkapslingsanläggningen). Kriticitet ska inte kunna vara en händelse som är aktuell för slutförvaret. I slutförvarsanläggningen ska det kunna visas att händelser med stor retardation/acceleration inte kan leda till kriticitet.”

Chapter 8, section 1.3.4 of SKBdoc 1091141 (reference 3) states: ”Kriticitetshändelser i slutförvaret kan därmed inte uppstå. Verifiering att denna dimensioneringsförutsättning för slutförvaret är uppfyllt redovisas i [4].” The [4] in this quote is a reference in that report: “[4] SKB 2009. Design premises for a KBS-3V repository based on results from the safety assessment SR-Can and some subsequent analyses, SKB TR-09-22”

3. PWR fuel specification[1]

3.1. PWR fuel types and specifications

A number of suppliers of nuclear fuels have been and will be used by the different nuclear power plants. The detailed design of the assemblies can vary between suppliers; see Appendix A in TR-10-13 (the Spent Fuel Report, reference 4). Appendix A contains information about existing PWR fuel types in Sweden at the end of 2008. There are minor variations not specified in that Appendix A but the main types are representative.

The information is repeated in Tables 1 and 2. Table 2 is based on Table A-4 of TR-10-13 (which refers to SKBdoc 1193244, reference 5). The selected information from TR-10-13 is assumed to be correct, except for some minor uncertainties and some editorial changes marked by the author in red in Table 2. The source documents for the information are referred to in Table 22 footnotes which are obtained from Appendix 3 of SKBdoc 1193244. Some of the specifications in Table 2 (from TR-10-13) are different to Table 22 (from SKBdoc 1193244) specifications and may be more accurate since they are results of further checks. The uncertain specifications are marked in red in both Tables 2 and 22.

[1] The text in this chapter is based on, often copied from, SKB source documents

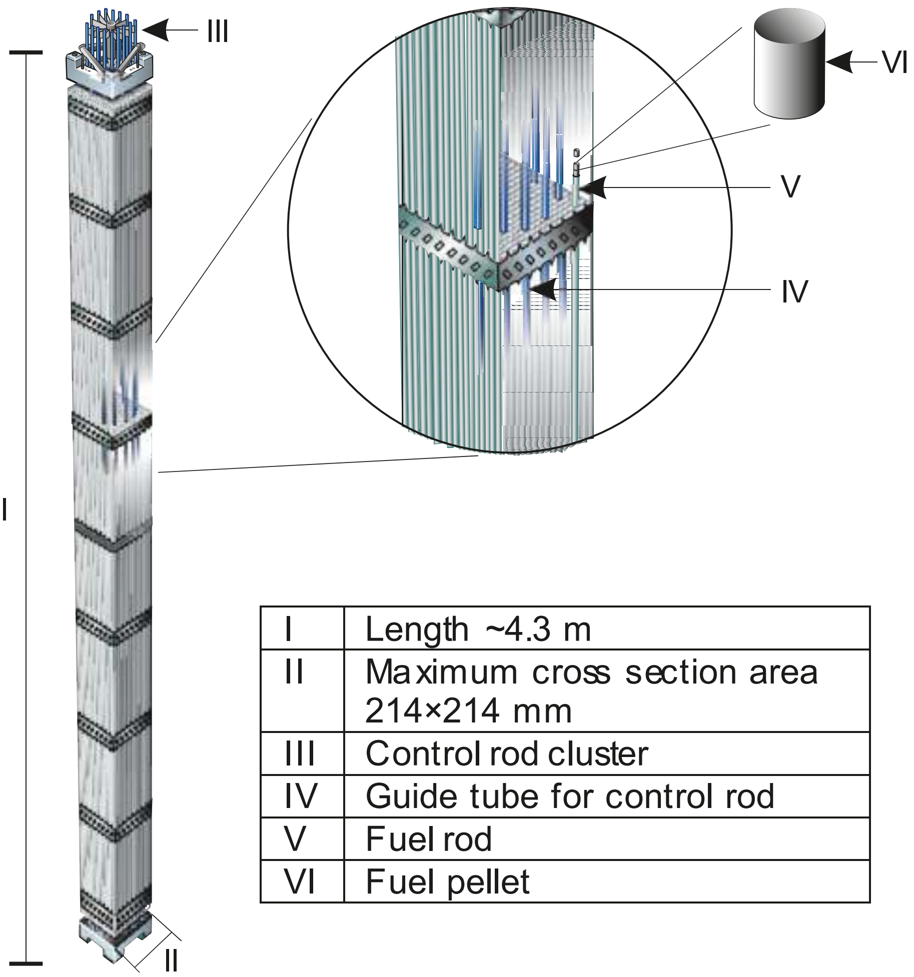
There are no geometry specifications for PWR MOX fuel available to the author during this review.

Table 3 contains design information on limiting fuel parameters that have been observed by SKB. The information is needed to design and test the inserts for the copper canisters.

Table 1: PWR fuel types (from Table A-1 of SKB TR-10-13)

PWR fuel type	Number of assemblies	Comment
W15×15	370	
Areva 15×15	170	
KWU 15×15	640	
W 17×17	520	
F 17×17	890	
AA 17×17	170	
F 17×17 AFA3G (incl. variants)	270	
17×17 HTP (incl. X5,M5)	330	More will come

"The PWR fuel assemblies contain 204 or 264 fuel rods, arranged in square arrays. The cross-sectional area is about 0.214×0.214 m^2 and the total length is about 4.3 m. Figure 1 shows a model of a PWR fuel assembly." (Section 2.3.2 of TR-10-13).



I	Length ~4.3 m
II	Maximum cross section area 214×214 mm
III	Control rod cluster
IV	Guide tube for control rod
V	Fuel rod
VI	Fuel pellet

Figure 1: A PWR fuel assembly. (SKB TR-10-13, Figure 2-5)

Table 2: PWR fuel type specifications (from Table A-4 of SKB TR-10-13).[1]

Fuel type	W15x15	KWU15x15	F15x15 AFA3G	15x15 AGORA	W17x17	AA17x17	F17x17	S17x17 HTP	17x17 HTPX5	17x17 HTP M5	17x17 HTP X5	17x17 AFA3G
No of fuel rods	204	204	204	204	264	264	264	264	264	264	264	264
Fuel rod pitch (mm)	14.3	14.3	14.3	14.3	12.6	12.6	12.6	12.6	12.6	12.6	12.6	12.6
Fuel rod outer diameter (mm)	10.72	10.75	10.72	10.77	9.5	9.5	9.5	9.55	9.55	9.5	9.5	9.5
Fuel rod inner diameter (mm)	9.48	9.3	9.484	9.505	8.36	8.36	8.36	8.33	8.33	8.364	8.35	8.355
Cladding thickness (mm)	0.62	0.725	0.618	0.6325	0.57	0.57	0.57	0.61	0.61	0.568	0.575	0.5725
Pellet diameter (mm)	9.20	9.11	9.294	9.33	8.19	8.19	8.19	8.17	8.165	8.192	8.192	8.192
Cladding material	Zr4	Zr4	M5	Zr4	Zr4	Zr4	Zr4	Zr4	Zr4	M5	M5	Zr4
Active fuel length (mm)	3658	3658	3658	3658	3658	3658	3658	3658	3658	3658	3658	3658
UO2 density (g/cc)*	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7	10.7
No of guide tubes	20	20	20	20	24	24	24	24	24	24	24	24
Guide tube material	Zr4	Zr4	M5	M5	Zr4	Zr4	Zr4	PCAm	PCAm	PCAm	PCAm	Zr4
Guide tube outer diameter (mm)	13.87	13.86	14.1	14.1	12.09	12.24	12.05	12.24	12.45	12.45	12.24	12.45
Guide tube inner diameter (mm)	13.01	13	13.05	13.05	11.05	11.44	11.25	11.3	11.45	11.45	11.3	11.45
Guide tube cladding thickness (mm)	0.43	0.43	0.525	0.525	0.52	0.4	0.4	0.47	0.5	0.5	0.47	0.5
No of instrument tubes	1	1	1		1	1	1	1	1	1	1	1
Instrument tube material	Zr4	Zr4	M5	M5	Zr4	Zr4	Zr4	PCAm	PCAm	PCAm	PCAm	Zr4
Instrument tube outer diameter (mm)	13.87	13.86	14.1	14.1	12.24	12.24	12.05	12.24	12.24	12.45	12.24	12.45
Instrument tube inner diameter (mm)	13.01	13.03	13.05	13.05	11.428	11.428	11.25	11.3	11.3	11.45	11.3	11.45
Instrument tube cladding thickness (mm)	0.43	0.43	0.525	0.525	0.406	0.406	0.4	0.47	0.47	0.5	0.47	0.5

* The UO2-density 10.7 g/cc is used for all fuel types. This density is higher than the fabricated values.

[1] Red text is assumed to be correct but data in Table A-4 in TR-10-13 and its source in Appendix 3 of SKBdoc 1193244 appear to be inconsistent with each other.

Table 3: "Design measures for the fuel channel tubes of the insert" (Table 3-1 in SKB TR-10-13)

"Detail	BWR (mm)	PWR (mm)	Comment
Longest assembly	4,441		Before irradiation
Induced length increase	14		When determining the length of the longest assembly the length before irradiation and the induced length increase is considered.
Largest cross section	141×141	214×214	Before irradiation.
Deviations due to deformations during operation	145.5×145.5	228×228	Cross sections of BWR transport cask, and PWR storage canister respectively. All assemblies in Clab have been placed in these casks or canisters, i.e. these cross sections are sufficient with respect to occurring deviations due to deformations during operation"

3.2. Unirradiated PWR fuel assembly contents

Typical contents of an unirradiated PWR fuel assembly are presented in Table 4. Other enrichments will give different weights. Typical impurities are presented in Table 5; structural components in the fuel assembly are presented in Table 6 while the control rod clusters are specified in Table 7. The information in Tables 5-7 is more extensive than what is needed for depletion and criticality safety assessment.

Table 4: Unirradiated fuel contents in a PWR fuel assembly (from Table B-1 of TR-10-13)

	PWR AREVA 17x17 4.0 (% U-235)	Weight in one fuel assembly (kg)
Fuel	U-tot	464
	U-234[1]	0.19
	U-235[1]	18.6
	U-236[1]	0.1
	U-238[1]	445.2
	O	62
Cladding material	Zirconium alloys	108
	Stainless steel	3
Other constructions (bottom and top plate, spacers etc.)	Stainless steel	12
	Zirconium alloys	21
	Nickel alloys	2

[1] In the fuel matrix a content of 0.04 wt.% ^{234}U + 0.02 wt.% ^{236}U for PWR is assumed.

Table 5: "Impurities in the fuel matrix" (Table B-2 of SKB TR-10-13)

Element	Assumed in calculations (ppm)	Representative values for fuel matrices[1] (ppm)	
Ag	0.05	<0.05	
L	6	3–6	
B	0.05	<0.05	
Bi	0.5	<0.5	
Ca	3	<3	1/3 above LRV
Cd	0.233	average 0.233	min 0.2 max 0.6
Co	0.5	<0.5	
Cr	1	<1	10% above LRV
Cu	0.5	average 0.5	min 0.2 max 7
F	2	<2	20% above LRV
Fe	5	<5	20% above LRV
In	0.3	<0.3	
Li	0.05	<0.05	
Mg	1	<1	
Mn	2	<2	
Mo	5	<5	
N[2]	14	–	
Ni	–	<1	
Pb	0.6	<0.6	20% above LRV
Si	10	<10	
Sn	0.8	0.6–0.8	
Ti	10	<10	
V	0.3	<0.3	
Zn	25	<25	
Dy	10	<10	
Eu	0.02	<0.02	
Gd	0.06	<0.06	
Sm	0.04	<0.04	
C	8.4	average 8.4	min 3 max 28
Cl	2	2	
Ni	5	5	
W	0.2	0.2	
(LRV_Lowest reported value)			

[1] Personal communication Westinghouse.

[2] Assumed in accordance with /SKBdoc 1179234/.

Table 6: PWR fuel assembly component materials (Table B-4 of SKB TR-10-13)

Compon-ent	Top nozzle SS	Top nozzle Zr	Bottom nozzle SS	Spacer Zr	Spacer inconel	Guide thimble Zr	Guide thimble SS	Cladding Zr	Cladding SS[1]
Material	304L_1	Inc718_1	304L_2	Zry4_3	Inc718_3	Zry4_4	316L_4	M5_5	302_5
Weight (kg)	6.5	1.1	4.8	7.2	0.75	14.2	0.7	108.1	3
Material Composition (%)									
Li	0.00001		0.00001				0.00001		0.00001
C	0.02	0.03	0.02	0.015	0.03	0.015	0.025	0.015	0.12
N	0.04	0.01	0.04	0.004	0.01	0.004	0.04	0.004	0.04
O	0.01	0.01	0.01	0.14	0.01	0.14	0.01	0.14	0.01
Na	0.001		0.001				0.001		0.001
Al	0.002	0.5	0.002	0.005	0.5	0.005	0.002	0.005	0.002
Si	0.6	0.3	0.6	0.01	0.3	0.01	0.6	0.01	0.6
P	0.02	0.005	0.02		0.005		0.02		0.02
S	0.015	0.005	0.015		0.005		0.015		0.015
Cl	0.0001	0.0001	0.0001		0.0001		0.0001		0.0001
Ca	0.002		0.002				0.002		0.002
Ti	0.01	0.9	0.01	0.004	0.9	0.004	0.01	0.004	0.01
V	0.001	0.01	0.001		0.01		0.001		0.001
Cr	18.5	19	18.5	0.1	19	0.1	17		18.5
Mn	1.3	0.3	1.3	0.003	0.3	0.003	1.3		1.3
Fe	69.05	21.05	69.05	0.22	21.05	0.22	66.25		69.95
Co	0.03	0.05	0.03	0.0001	0.05	0.0001	0.03	0.00001	0.03
Ni	10	52.5	10	0.004	52.5	0.004	12		9
Cu	0.1	0.1	0.1	0.003	0.1	0.003	0.1		0.1
Zn	0.01		0.01				0.01		0.01
As	0.01	0.05	0.01		0.05		0.01		0.01
Se	0.004		0.004				0.004		0.004
Nb	0.03	4.59	0.03	0.01	4.59	0.01	0.03	1	0.03
Mo	0.2	0.05	0.2	0.0005	0.05	0.0005	2.5		0.2
Ag	0.0001		0.0001				0.0001		0.0001
Sn	0.01	0.01	0.01	1.5	0.01	1.5	0.01		0.01
Sb	0.001	0.005	0.001		0.005		0.001		0.001
Ce	0.01		0.01				0.01		0.01
Ta	0.01	0.51	0.01		0.51		0.01		0.01
W	0.01	0.01	0.01	0.005	0.01	0.005	0.01	0.005	0.01
Hf				0.006		0.006		0.01	
Zr	0.001			97.97		97.97		98.81	
Th				0.00002		0.00002		0.00002	
U				0.00015		0.00015		0.00015	

[1] The stainless steel in the springs in the fission gas plenum.

Table 7: PWR fuel assembly control rod cluster contents (Table B-5 of SKB TR-10-13)

Component	Absorber pins	Absorber pins	Top piece
Material	304_1	AgInCd_1	304_2
Weight (kg)	12.5	51.4	3.8
Material composition (%)			
Li	0.00001		0.00001
C	0.07		0.07
N	0.04		0.04
O	0.01		0.01
Na	0.001		0.001
Al	0.002		0.002
Si	0.6		0.6
P	0.02		0.02
S	0.015		0.015
Cl	0.0001		0.0001
Ca	0.002		0.002
Ti	0.01		0.01
V	0.001		0.001
Cr	18.5		18.5
Mn	1.3		1.3
Fe	68.9		68.9
Co	0.1		0.1
Ni	10		10
Cu	0.1		0.1
Zn	0.01		0.01
As	0.01		0.01
Se	0.004		0.004
Nb	0.03		0.03
Mo	0.2		0.2
Ag	0.0001	80	0.0001
Sn		5	
Sb		15	
Ce	0.01		0.01
Ta	0.001		0.001
W	0.01		0.01
Hf	0.01		0.01
Zr	0.01		0.01

Table 8 shows distributions of PWR canisters to be disposed of per year and canister type. Green background refers to 1057 PWR I type canisters. Red background refers to 38 PWR II type canisters and grey background to 557 PWR III type canisters. A canister type specifies a range of burnup values and decay times of the fuel in the canister. More information is provided in chapter 3.7 of this Technical Note.

Table 8: PWR canisters/year during 2023–2070 (PWR data from Table C-4 of SKB TR-10-13)

Year	Burnup range →	<42 MWd			42–47 MWd			>47 MWd		
	No of assemblies →	4	3	<3	4	3	<3	4	3	<3
2023	6	6								
2024	22	22								
2025	33	33								
2026	33	33								
2027	33	33								
2028	41	41								
2029	41	39			2					
2030	41	37			4					
2031	41	32			9					
2032	41	25			16					
2033	41	22			19					
2034	41	10			31					
2035	41	4			37					
2036	41	1			40					
2037	41	1			40					
2038	41	2			39					
2039	41				41					
2040	41				41					
2041	41				41					
2042	41				41					
2043	41				39			2		
2044	41				40			1		
2045	41				31			10		
2046	41				31			10		
2047	41				37			4		
2048	41				31			10		
2049	41				31			1	9	
2050	41								41	
2051	41	32							9	
2052	41								41	
2053	41	18				23				
2054	41					2			39	
2055	27								27	
2056	27								27	
2057	27								27	
2058	27								27	
2059	27								27	
2060	27								27	
2061	27								27	
2062	27								27	
2063	27								27	
2064	27								27	
2065	27								27	
2066	27								27	
2067	27								27	
2068	27				1				26	
2069	27				12				15	
2070	13				12				1	
Total	1,652	391	0	0	666[1]	25	0	38	532	0

[1] Including 33 PWR MOX assemblies

3.3. Depleted PWR fuel assemblies

The maximum assembly average burnup for PWR fuel with uranium oxide (UOX) fuel is specified as 60 MWd/kgU (TR-10-13 Section 2.1.1).

The following PWR UOX data are obtained from Table 2-2 in TR-10-13:

PWR UOX	
Year 2045 from the reactors R2, R3 and R4	6,016 assemblies
31 December 2007 from R2, R3 and R4	2,552 assemblies
Assumed weight U per assembly:	464 kg

The following PWR MOX data are obtained from Table 2-2 and text on swap MOX fuel in Section 2.2.2 in TR-10-13:

PWR MOX	
Swap from Germany	33 assemblies
Initial mass of actinides:	8.4 ton.
The average burnup is:	31 MWd/kg initial actinides

The following two quotes are from the second paragraph in Section 2.2.1 in TR-10-13:

“Approximately one out of four of the PWR assemblies will contain a control rods cluster”.

“Since the beginning of 1970 the burnup of the nuclear fuel has increased from approximately 23 MWd/kgU up to 53 MWd/kgU. The average burnup of the PWR fuel stored in the interim storage facility is about 41 MWd/kgU (December 2007). For the remaining operation, the burnup will increase as a result of increased power and optimisation of the operation of the reactors.”

The third paragraph of the same Section states that the resulting average burnup for the reference scenario is 44.8 MWd/kgU for PWR fuel assemblies (a reference to SKBdoc 1221579 is made).

The burnup distribution of PWR fuel stored in Clab on 31 December 2007 and a prognosis for future PWR fuel burnup values are presented in Figure 2.

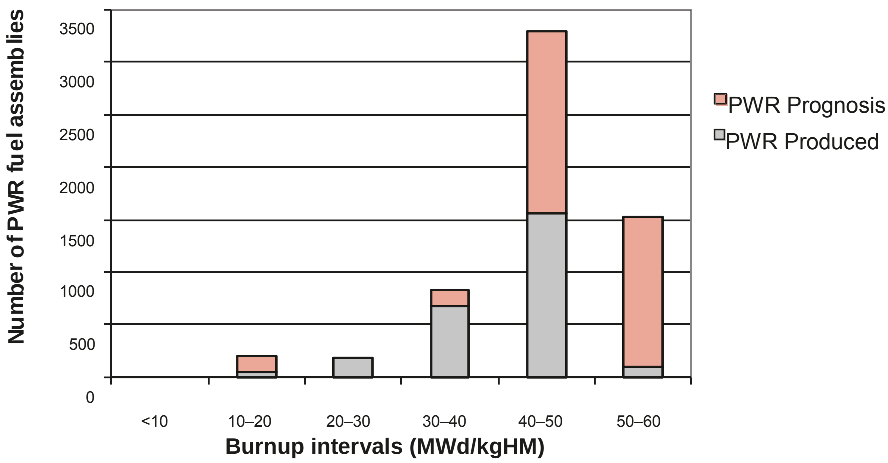


Figure 2: PWR fuel assembly burnup distributions (SKB TR-10-13, Figure 2-1)

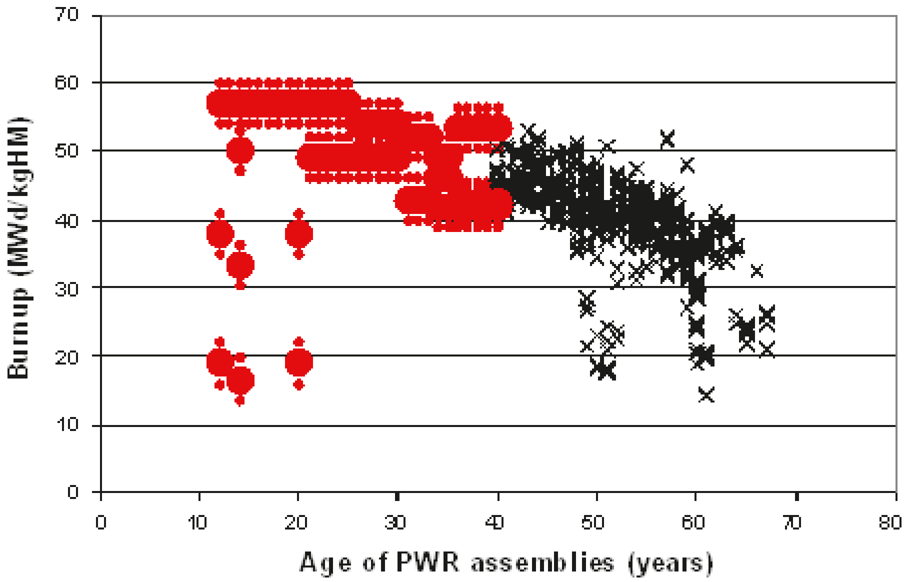


Figure 3: PWR fuel assembly ages. (SKB TR-10-13, Figure 2-4)

In Figure 3 the PWR "assemblies stored in Clab at the end of 2007 and the assemblies included in the reference scenario and their burnup and age are plotted for 2045, i.e. the last year of operation of the last reactor to close down". "For the assemblies included in the reference scenario large red dots represent the batch average discharge burnup. The smaller red dots represent the assumed standard deviation in burnup, i.e. ±3 MWd/kgU, averaged over single fuel assemblies included in a batch. Each dot represents several assemblies. Low burnup assemblies in the reference scenario are from the last year of operation of the nuclear power plants." (The above quotes are from the text preceding Table 2-2 in TR-10-13)

3.4. Selection of PWR fuel assemblies for canister

Criticality safety related requirements and criteria for the selection of fuel assemblies to be encapsulated include the following:

- "Requirement on handling: *The fuel assemblies to be encapsulated shall be selected with respect to enrichment, burnup, geometrical configuration and materials in the canister so that criticality will not occur during the handling and storage, even if the canister is filled with water*.
- Criterion: *The effective multiplication factor (k_{eff}) must not exceed 0.95 including uncertainties.*"
  (from Section 3.1.2 in TR-10-13)
- Requirement on handling: *Before the fuel assemblies are placed in the canister they shall be dried so that it can be justified that the allowed amount of water stated as a design premise for the canister is not exceeded.*
- Criterion: *The amount of water left in anyone canister shall be less than 600 g.* (from Section 3.1.4 in TR-10-13)

- "Requirement on handling: *Before the canister is finally sealed, the atmosphere in the insert shall be changed so that acceptable chemical conditions can be ensured.*
- Criterion: *The atmosphere in canister insert shall consist of at least 90% argon.*"
  (From Section 3.1.4 in TR-10-13)
- "Requirement on handling: *The number of canisters shall be minimised and, if possible, all assembly positions in the deposited canisters shall be filled.*"
  (from Section 3.2.1 in TR-10-13)

"The selection process can be summarised as follows.

1. Compile information for the selection.
2. Preliminarily selection – based on decay power and the objective to fill all assembly positions in the canisters to be deposited.
3. Check criticality – adjust the selection in case of non-conformity to the criterion for criticality.
4. Check radiation dose rate on the canister surface – adjust the selection in case of non-conformity to the criterion for maximum allowed radiation dose rate.
5. Lifts and movements – investigate the number of lifts and movements of assemblies and storage canisters and adjust the selection if the number of lifts can be reduced and the selection still conforms to criteria for decay power and criticality.
6. Final selection – determine a selection and make a plan for transport of storage canisters and assemblies."
   (From Section 4.1.1 in TR-10-13)

"Criticality safety is checked by calculating loading curves" (text after Figure 4-2 in TR-10-13). "The calculated loading curves and the combinations of average burnup and enrichment for the assemblies that currently are stored in Clab are given in Figure" 4. "Fuel assemblies with a combination burnup/enrichment that are plotted above the loading curves in Figure "4 "may result in a canister k_{eff} that exceeds 0.95 and will thus not conform to the general criterion for criticality" safety (quotes from paragraph before Figure 4-3 in TR-10-13).

3.5. Burnup determination and assembly selection

The process of selecting appropriate fuel assemblies for encapsulation is extremely important for safety and efficiency. The following quote from Section 4.2.2 of TR-10-13 is descriptive: "Regarding the burnup SKB intends to use data provided by the nuclear power plants. The burnup must be regularly calculated and accurately measured to achieve a reliable and efficient operation of the reactors. The registration of burnup during operation is required by the Swedish Radiation safety authority (SSM) and the data is quality assured. If required, the burnup can also be measured by γ-scanning."

3.6. Burnup loading curves

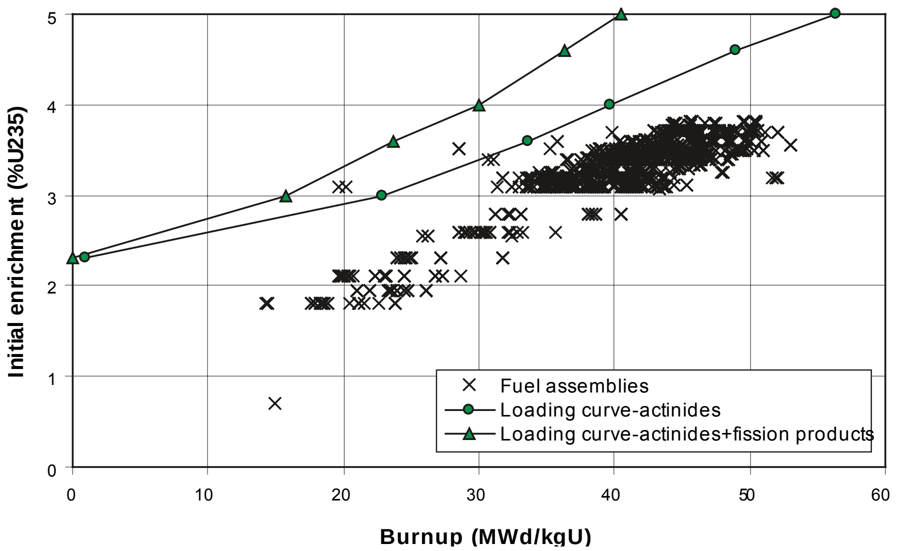


Figure 4: Loading curves for PWR-canisters for assemblies currently stored in Clab (SKB TR-10-13, Figure 4-4).

The following three paragraphs are based on text in Section 4.4.1 of TR-10-13 under "Check of criticality".

After the primary selection based on the decay power has been made, the criteria to avoid criticality are checked. If $k_{eff} < 0.95$ for canisters[3] with identical assemblies of each type to be stored in the canister, i.e. the combination of enrichment and burnup, lies under the calculated loading curve, the assemblies can be encapsulated without further checks.

There will be canisters with $k_{eff} > 0.95$ if loaded with identical assemblies having some specific characteristics. Such assemblies will not comply with the loading curve. In such cases, the k_{eff} value may be calculated for the actual set of selected assemblies. In these calculations, an assembly potentially causing a $k_{eff} > 0.95$ is placed in the canister in the worst position for potential criticality. If the calculations show that k_{eff} is still above 0.95 for the canister, that selection of assemblies is not encapsulated.

If it is not possible to find a set of assemblies that conform to the general criticality safety criteria, a low burnup assembly can be encapsulated alone in a canister. Should it neither be possible to combine the low burnup assemblies with high burnup assemblies nor to encapsulate such assemblies individually to conform to the criticality criteria, the ultimate measure is to alter the geometry, i.e. to reconstruct the assembly.

[3] TR-10-13 refers to assemblies rather than canisters with k_{eff} related to 0.95

3.7. Canister types I, II and III

Three types of canisters for PWR UO_2 fuel have been used by SKB to select fuel assemblies for encapsulation. Canister data are provided in Table 8 as well as in Figure 5 and Figure 6.

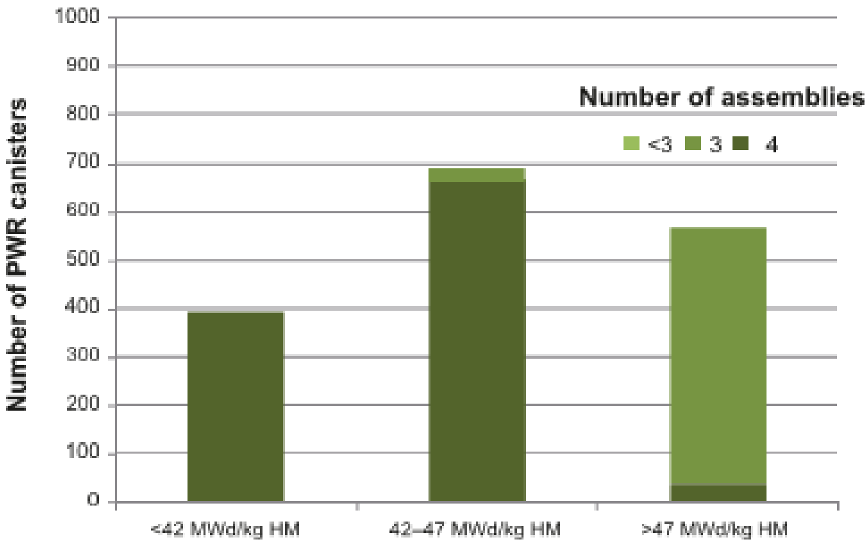


Figure 5: PWR canisters in groups with average burnup and assemblies per canister shown (SKB TR-10-13, Figure 5-3).

SKB has identified a total of eight types of canisters, four of each for PWR (PWR MOX fuel is added to the UO_2 fuels) and BWR fuel assemblies, as shown in Figure 6. According to Section 6.2.4 of TR-10-13, the selection of type canisters is made based on the burnup of the assemblies since it is the main parameter determining the radionuclide inventory:

- The PWR I canister has been selected since it represents the average canister resulting from the simulation of the encapsulation of the assemblies to be deposited.
- The high burnup PWR II canister has been selected since it represents the high end canisters with respect to radionuclide inventory.
- The PWR III canister has been selected to represent the unfilled canisters, which are the result of the current decay power criterion and assumed encapsulation period.
- Finally, the PWR MOX canister has been selected to represent the canisters containing PWR MOX assemblies.

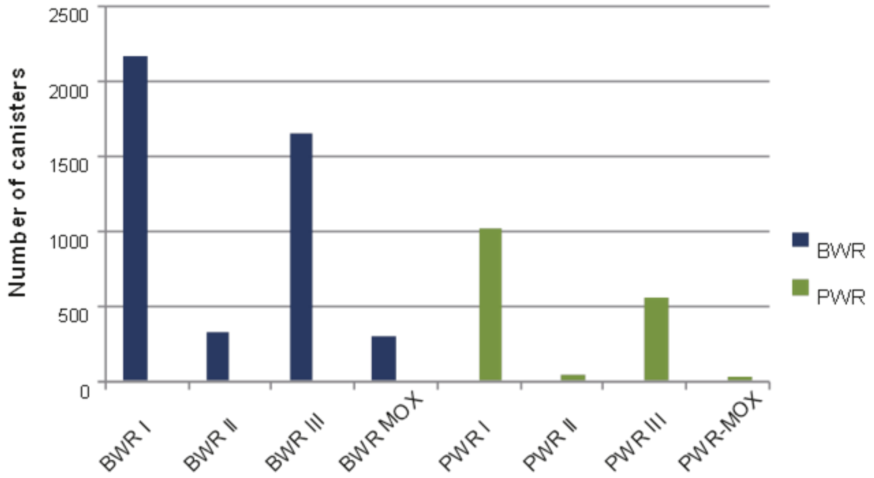


Figure 6: BWR and PWR canister inventory per canister type (SKB TR-10-13, Figure 6-1)

3.7.1. The PWR I type-canister

The following quotes are from Section 6.2.4 in TR-10-13:

"The average burnup of the assemblies in this canister is 44.8 MWd/kgU and the radionuclide inventory is regarded to be representative for all canisters where four PWR assemblies with different burnup and age have been combined so that their total decay power is 1,700 W and their average burnup lies in the interval 42–47 MWd/kgU. The PWR I type canister also represents full canisters with an average burnup of the assemblies less than 42 MWd/kgU. Assuming the same radionuclide inventory in these canisters as in the PWR average canister will result in an overestimated but still adequate inventory."

The age of each assembly is at least 38 years (Table C-7 in TR-10-13).

A summary of the canisters, for which the PWR I canister is considered to provide an adequate description of the radionuclide inventory, is given in Table 9.

Table 9: PWR canister type I statistics (Table 6-9 in SKB TR-10-13)

	Radioactive inventory		
	Representative	Overestimated but adequate	Total
Number of canisters	633	391	1,024
Part of PWR canisters (1,652)	38%	24%	62%
Part of all canisters (6,110)	10%	6%	17%

3.7.2. The PWR II type-canister

The following quotes are from Section 6.2.4 in TR-10-13:

“The radionuclide inventory in the PWR II type-canister is set to the inventory in the canister denominated “PWR high burnup”. “The average burnup of the assemblies in this canister is 57 MWd/kgU. With respect to the total inventory of assemblies to be deposited, the applied criteria for selection of assemblies and the assumed period for encapsulation and deposition, the radionuclide inventory in the PWR II canister is regarded as the high end of the PWR canisters to be deposited”.

”The radionuclide inventory in the PWR II canister represents all canisters where PWR assemblies with different burnup and age have been combined so that their total decay power is 1,700 W and their average burnup is at least 45 MWd/kgU. In most of these canisters, the average burnup of the assemblies will be less than 57 MWd/kg/U. A summary of the canisters, for which the PWR II canister is considered to provide an adequate description of the radionuclide inventory, is given in Table” 10.

The age of each assembly is at least 55 years (Table C-11 in TR-10-13).

Table 10: PWR canister type II statistics (Table 6-10 in SKB TR-10-13)

	Representative radionuclide inventory
Number of canisters	38
Part of PWR canisters (1,652)	2%
Part of all canisters (6,110)	1%

3.7.3. The PWR III type canister

The following quotes are from Section 6.2.4 in TR-10-13:

“The radionuclide inventory in the PWR III type-canister is set to the inventory in the canister denominated “PWR combination b””. “The PWR III canister represents all PWR canisters with three assemblies. Based on the results from the simulation of the encapsulation, there are no PWR canisters with less than three assemblies. The average burnup of the assemblies in this canister is 57 MWd/kgU. The bulk of the canisters with three assemblies will have an average burnup lower than this.

The content of short lived fission and activation products mainly depends on the burnup and age of the assemblies and will be similar as for the full canisters. The content of transuranium elements and isotopes with long half-lives will mainly depend on the encapsulated mass of uranium and will, thus, be lower than in the canisters that contain four assemblies. The number of canisters containing three assemblies is given in Table” 11.

One of the PWR assemblies has an age of at least 20 years while the other two assemblies have ages of at least 51 years (Table C-12 in TR-10-13).

Table 11: PWR Canister type III statistics (Table 6-11 in SKB TR-10-13)

	Representative radionuclide inventory and total
Number of canisters	557
Part of PWR canisters (1,652)	34%
Part of all canisters (6,110)	9%

3.7.4. The PWR-MOX canister

The following quotes are from Section 6.2.4 in TR-10-13:

"The radionuclide inventory in the PWR-MOX type-canister is set to the inventory in the canister denominated "PWR-MOX"". "Each PWR-MOX canister contains one swap PWR MOX assembly. The burnup of the MOX assembly is set to 34.8 MWd/kg actinides, which is the maximum burnup of the swap PWR MOX assemblies. The burnup of the remaining three assemblies in the canister is 44.8 MWd/kgU, i.e. the average PWR assembly burnup. The radionuclide inventory in the PWR-MOX canister is regarded to be representative for all PWR canisters containing a MOX assembly.

A summary of the canisters, for which the PWR-MOX canister is considered to provide an adequate description of the radionuclide inventory, is given in Table" 12.

The age of the MOX assembly is at least 57 years and the age of each UO_2 assembly is at least 32 years (Table C-13 in TR-10-13).

Table 12: PWR MOX-canister statistics (Table 6-12 in SKB TR-10-13)

	Representative radionuclide inventory
Number of canisters	33
Part of PWR canisters (1,652)	2%
Part of all canisters (6,110)	1%

4. Reactor operating data for the fuel[4]

The Spent fuel report (TR-10-13) is based on a reference scenario for the future operation of the nuclear power plants and also includes the spent fuel that is stored in the interim storage facility. Alternative scenarios for the operation of the nuclear power plants are not included.

The batch average discharge burnup values for the ten remaining Swedish nuclear power plants that are assumed in the reference scenario used by SKB are presented in SKBdoc 1219727, ver 2.0. ("Confidential information. Available only for the Swedish Radiation Safety Authority.")

[4] The text in this chapter is based on, often copied from, SKB source documents

Some reactor operating data and typical axial profiles are specified in SKBdoc 1193244. Some of that information is quoted later in chapter 8 of this Technical Note. Table 13 provides some reactor data that may be of relevance (e.g. increased power) to criticality safety.

At this time it is not clear to the author what information will be useful to assess and control criticality safety for the copper canister. The need for detail will depend on the need to have a small subcritical margin (the safety margin is a different issue, always need to be significant). Some information on relationships between reactor operation and fission gas release may be of value. Section 6.3 in TR-10-13 provides information that is summarised below. Figure 7 and Figure 8 provide some data on fission gap release.

Table 13: Thermal reactor power and last year of operation for the Swedish PWRs (From Table 2-1 in SKB TR-10-13)

Reactor	Reactor power (MW_{th})	Increases in reactor power (MW_{th}) 2009	2012	Last year of operation
R2	2,652			2025
R3	2,992	3144		2031
R4	2,775		3300	2033

The power history, i.e. the power developed per length unit of fuel rod or fuel assembly over the irradiation period in the reactor vessel, referred to as the linear heat generation rate, is strongly correlated to the fission gas release (FGR). The FGR in turn is used to determine the part of the radionuclide inventory that is located at the fuel grain boundaries and in the gap between the fuel and the cladding. This part of the inventory is referred to as the gap inventory and will in comparison to the radionuclides embedded in the fuel matrix be released very rapidly if the spent fuel pellets are exposed to vapour or water.

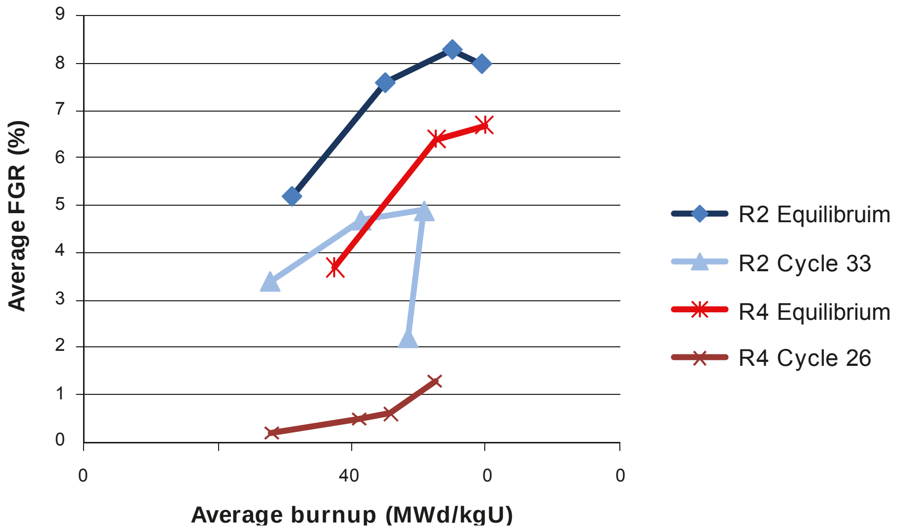


Figure 7: "Calculated average fission gas release at the end of each cycle for PWR cases" (SKB TR-10-13, Figure 6-3).

The relations illustrated in Figure 7 "are used to extrapolate reactor-specific relations between average burnup and FGR. In the interpolation, the numbers of assemblies in the reactors and their thermal powers have been considered. The relations are based on the assumption that the FGR is correlated to the linear heat generation. The interpolated relations between burnup and FGR have then been used to estimate the FGR of the spent fuel assemblies included in the reference scenario for the operation of the nuclear power plants" (from text following Figure 6-3). "The drop in burnup and FGR for R2 (Ringhals 2) Cycle 33 is explained by that only the low burnup assemblies were loaded in the last cycle." (From Figure 6-3 title).

The following quotes are from Section 6.3.1 of TR-10-13:

"From the average burnup of each assembly, the reactor it has been used in, and whether it was used before or after the increase in power, the extrapolated reactor-specific relations were used to estimate the FGR in each individual assembly."

"The resulting average FGR for all PWR assemblies is 4.3% with a standard deviation of ±3.11%. The number of PWR assemblies in different FGR intervals is illustrated in Figure" 8

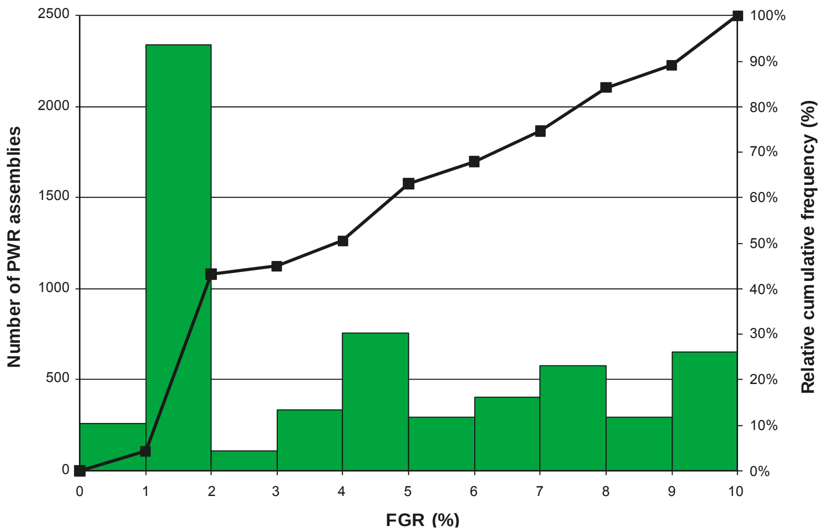


Figure 8: PWR assemblies in FGR intervals and FGR relative cumulative frequency (SKB TR-10-13, Figure 6-5)

"For the PWR-MOX canister the FGR was not estimated since the information required to estimate reactor-specific relations between burnup and FGR was not available for the German reactors from which the PWR-MOX assemblies originate. With respect to the low burnup of the MOX assembly and an average burnup close to that of the PWR I canister, the FGR in these canisters can be assumed to be similar or less than in the PWR I type canister." (From Section 6.3.2 of TR-10-13).

5. The copper canister specifications[5]

The specifications for the copper canister are expected to be provided in a general part of the safety documentation, not only in the criticality safety part. In the SKB documentation for the application, this information appears to be compiled mainly in the Canister production report (TR-10-14, reference 6) and in the Data report (TR-10-52, reference 7).

The maximum total weight of the canister, including fuel, is 26,800 kg for PWR, see Table 14.

Table 14: "Weight of the canisters" (Table 3-1 in SKB TR-10-14).

PWR fuel type	Weight (kg)	
	BWR-canister	PWR-canister
Insert with lid	13,700	16,400
Copper shell	7,500	7,500
Canister without fuel	21,200	23,900
Canister with fuel	24,600–24,700	26,500–26,800

"The canister comprises the following components which are detailed in the following sections: cast iron insert with steel tube cassette, steel lid, copper tube, copper lid and copper base, see Figure" 9 (quoted from Section 3 in TR-10-14).

"the copper shell, i.e. tube, lid and base, are made of highly pure copper. The copper components are welded together by friction stir welding (FSW). To facilitate handling of the canister, the copper lid is provided with a flange to allow handling equipment to grip the canister." (Section 3.2 in TR-10-14). "The insert is manufactured of nodular cast iron with steel channel tubes in which the fuel assemblies are to be positioned." (Section 3.1 in TR-10-14).

The following quotes from Section 3 of TR-10-14 are selected: "The reference canister design comprises" one insert "for 4 PWR fuel assemblies". "The reference design is described by a set of design parameters for which nominal values and acceptable variations are given."

"The initial state of the canister is defined as the state when the canister is finally deposited" (first paragraph in Section 7 of TR-10-14) in the repository.

- "Design premise: *The spent fuel properties and geometrical arrangement in the canister should be such that criticality is avoided even if water should enter a canister.*" (Section 2.4.1 of TR-10-14)
- "Requirement on the handling: *The fuel assemblies to be encapsulated shall be selected with respect to enrichment, burnup, geometrical configuration and materials in the canister, so that criticality will not occur during the handling and storage, even if the canister is filled with water.*" (Section 2.4.1 of TR-10-14)

[5] The text in this chapter is based on, often copied from, SKB source documents

Table 15 shows the material compositions of the copper shell (the copper itself is not shown but is more than 99 % by mass) and of the insert.

Table 15: "Material composition at the initial state" (from Table 7-1 of SKB TR-10-14)

Component	Design parameter	Reference design	Initial state value
Insert	Copper content	<0.05	<0.05
	Iron content	>90	>90
	Carbon content	<4.5	<4.5
	Silicon content	<6	<6
Copper shell	Phosphorus (ppm)	30–100	30–100
	Sulphur (ppm)	<12	<12
	Hydrogen (ppm)	<0.6	<0.6
	Oxygen (ppm)		
	- Tube	up to some tens	<5
	- Lid and base	up to some tens	<5
	- Weld	up to some tens	up to some tens

The reference canister for BWR fuel is shown in Figure 9 and Figure 10 (similar presentations of a PWR canister have not been found).

For the copper shell, Section 3.2.3 of TR-10-14 states: "The dimensions are given in the figures and tables below. All dimensions are specified at room temperature, 20°C." The figures 3-3, 3-7, 3-8 and 3-9 correspond to figures 10-13 here. There is only one table (3-6) in TR-10-14 and it is quoted in Table 16 here.

"The dimensions of the cast iron insert with the steel lid are given in the figures and tables below. All dimensions are specified at room temperature, 20°C." The quote is from Section 3.1.3 of TR-10-14 where the figures 3-4, 3-5 and 3-6 correspond to figures 14-16 here. Only a selection of TR-10-14 Table 7-3 data is quoted in Table 17 here (the tables referred to in the quote are not used).

Information on inserts provided in Table 3-3 and Table 3-5 of TR-10-14 (where a further reference to SKBdoc 1203875 is given) is merged into Table 18 here.

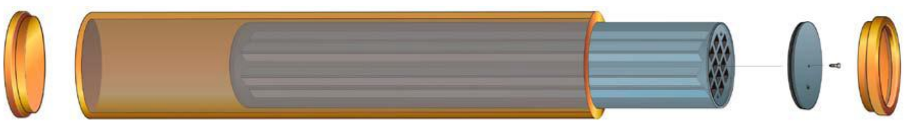


Figure 9: "Exploded view of the reference canister and its components (from the left: copper base, copper tube, insert, steel lid for insert and copper lid)" (SKB TR-10-14, Figure 3-3).

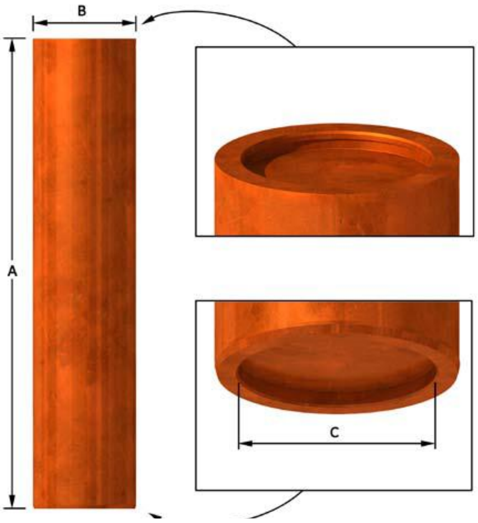
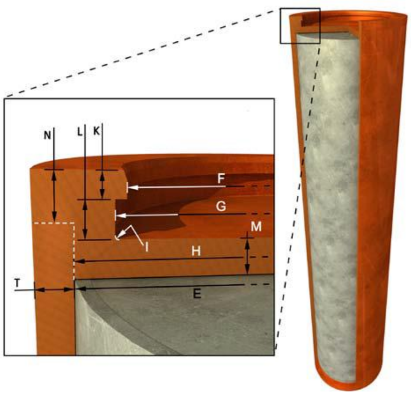


Figure 10: Reference canister (BWR fuel)
(SKB TR-10-14, Figure 3-1)

Figure 11: Copper shell dim., see Table 17
(SKB TR-10-14, Figure 3-7)



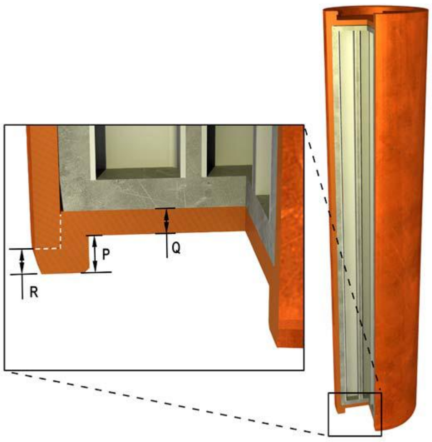


Figure 12: Copper lid dim., see Table 17
(SKB TR-10-14, Figure 3-8)

Figure 13: Copper base dimensions, see Table 17
(SKB TR-10-14, Figure 3-9)

Table 16: "Dimensions for copper shell /SKBdoc 1203875/" (Table 3-6 in SKB TR-10-14).

Figure no and dimension designation	Designation	Nominal value (mm)	Tolerance (mm)
11 A	Total length	4,835	+3.25/–2.75
11 B	Outer diameter	1,050	+1.2/–1.2
11 C	Inner diameter	850	+0.8/–0.8
12 T	Wall thickness Weld thickness*	49 48.5	+0.3/–0.3 +0.7/–0.7
12 E	Inner diameter	952	+0.5/–0.5
12 F	Inner diameter	821	+0/–0.5
12 G	Inner diameter	850	+0.8/–0.8
12 H	Diameter, lid	953	d8
12 H	Diameter, tube	953	H8
12 I	Corner radius	10	–
12 K	Dimension	35	+0.5/–0.5
12 L	Dimension	50	+0.2/–0.2
12 M	Thickness, lid	50	+0.6/–0.6
12 N	FSW position	top	60
13 P	Dimension	75	+0.3/–0.3
13 Q	Thickness, base	50	+1/–1
13 R	FSW position	50	
Calculated	Inner free length	4,575	+0.6/–0.1
Calculated	Axial gap between steel and copper lids	2	+1.1/–0.3
Calculated	Radial gap between shell and insert	1.5	+0.25/–0.5

* The weld thickness differs from the wall thickness since the copper tube surfaces that connect to the lid and base respectively are further machined.

Table 17: "Dimensions at initial state" (selection from Table 7-3 of SKB TR-10-14)

Component	Design parameter	Reference design	Initial state value	
Insert PWR	Edge distance (mm)	37.3 ± 10	37.3 ± 5	
	C-C distance between compartments (mm)	370 ± 3.6	370 ± 3.6[2]	
Copper shell	Thickness (mm)		All shell parts	Fraction of canisters
	- Tube	49.0	Minimum > 47.5:	> 99%
	- Lid and base	50.0	45–47.5[1]:	Few per thousand
	- Weld	48.5	Minimum < 45[1]:	Negligible
	- Local reduction due to defects	-	< 10 10–20[1] >20[1]	> 99.9% one per thousand negligible

[1] Values occurring only at disturbed operations considering both the manufacturing processes and inspection

[2] The initial state values are based on measures from the reference design

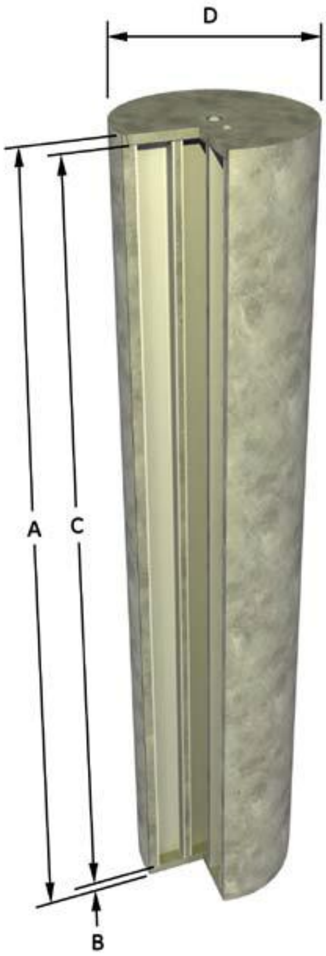


Figure 14: Insert dimensions. (SKB TR-10-14, Figure 3-4).

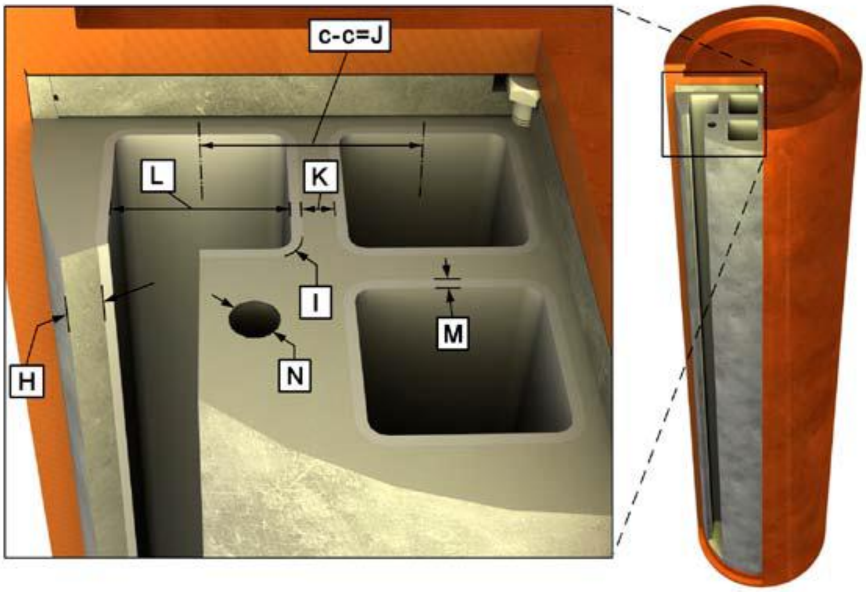


Figure 15: Insert (BWR) channel tubes with dimensions for PWR fuel, see Table 18 (SKB TR-10-14, Figure 3-5)

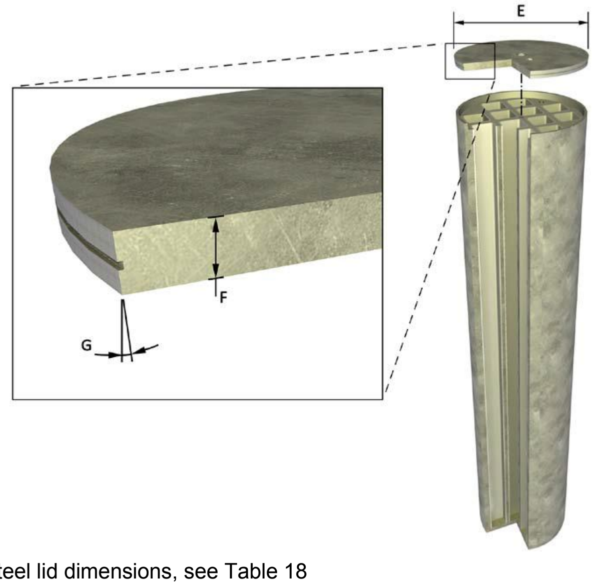


Figure 16: Steel lid dimensions, see Table 18 (SKB TR-10-14, Figure 3-6)

The inserts and the range of geometry variations that may be credible need to be specified and verified to an extent consistent with what will be credited for in the criticality safety assessment. SKB presents considerable information on this but the information is not complete and that is clarified in the documentation. Most of the verification and testing has involved BWR canisters with inserts. Figure 17 and Figure 18 show BWR canister inserts and some of the testing. Table 19 shows material checks while Table 20 shows measured dimensions for some PWR-inserts. Table 21 contains detailed information on the copper material used to form canister shells during 2005-2008.

Figure 17: Insert channel tube cassette (BWR) (SKB TR-10-14, Figure 5-4)

“Material properties for the insert steel lid are based on the steel S355J2G3” (Section 4.2.1 in TR-10-14).

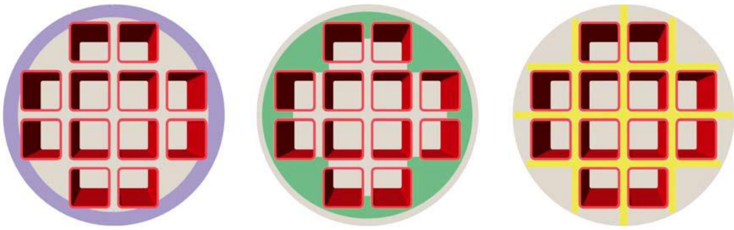


Figure 18: “Testing areas for BWR insert. The areas investigated with the various methods are angle scanning (lilac), normal scanning (green) and transmission testing (yellow)” (SKB TR-10-14, Figure 5-5)

Table 18: Dimensions for PWR-inserts.

Figure no and dimension designation	Designation	Nominal value (mm)	Tolerance (mm)
	Insert dimensions		
14 A	Length of insert	4,573	+0/–0.5
14 D	Insert diameter	949	+0.5/–0
	Steel lids		
16 E	Diameter	910	h7
16 F	Lid thickness	50	+0.1/–0.1
16 G	Bevel angle	5°	+0.1°/–0.1°
	Inserts		
14 B	Thickness of bottom	80	+10.1/–5.6
14 C	Interior length	4,443	+5/–10
15 H	Edge distance	37.3	+10/–10
15 N	Lifting eye holes	Two holes with M45	
	PWR-Insert channel tubes		
3-5 I	Ext. channel tube corner radius	20	+5/–5
3-5 K	Distance between channel tubes	110	+6.2/–6.2
3-5 J Calculated	C-C distance between compartments	370	+3.6/–3.6
3-5 L Calculated	Int. channel tube (before casting)	235×235	+5.1/–5.1*
3-5 M	Channel tube thickness	12,5	+1.25/–1.25
3-5	Ext. channel tube cross section	260	+2.6/–2.6

* This tolerance of inner cross section of channel tube is valid before casting.

Insert dimensions

A selection of text from Section 7.1.2 of TR-10-14 follows: “The specified edge distance is 33.3 ± 10 mm, giving an acceptable minimum measure of the edge distance of 23.3 mm.” “The results from the test manufacturing” “shows that manufactured inserts conform to the specification (misalignment of 3–8 mm). When considering the actions recently performed to reduce the misalignment of the cassette and the ultrasonic measurement, the misalignment under normal production can be assumed to be ± 5 mm. The probability to exceed the specified ± 10 mm is regarded to be negligible”.

Concerning dimensions for the internal channel tube the following information is obtained from Table 4-8 of TR-10-14: The reference design internal channel tube cross section before casting is 235 mm × 235 mm. Gauge dimensions (used after casting) are 226 mm × 226 mm (preliminary data). Section 7.1.6 of TR-10-14 specifies that: ”So far, no verification of the C-C distance between compartments by physical measurement has been done on manufactured inserts.”

Section 5.2.10 of TR-10-14 presents some testing results for manufacturing: “The development of PWR inserts had until 2007 been carried out on a significantly smaller scale. Subsequently, development has been intensified and, as a consequence of the experience gained in the manufacturing of BWR inserts, good progress has been made."

"The reported results are based on the five BWR inserts manufactured in 2007 and on the three PWR inserts manufactured with the channel tube dimension specified in the reference design."

Information from the dimension inspection is presented in SKBdoc 1175208 (the Manufacturing Report). Section 5.2.10 of TR-10-14 has the following text: "For PWR inserts, problems have been experienced in gauging with a gauge measuring 226×226 mm in size. For example, only one of these three inserts has met the gauge values. During 2007, the technique of inserting compacted sand into the channels before casting was further developed. When an improved compaction of sand has been used, it has been possible for the channels to be gauged after casting. The problem is now deemed to have been solved, but further means for improvement will be tested."

Table 19: "Three individually manufactured PWR inserts compared to the technical specifications for test manufacturing" (from Table 5-1 of SKB TR-10-14)

Material	Material composition for nodular cast iron (%) The content of Fe is above 90% in all inserts.							
	Cu	C	Si	Mn	P	S	Ni	Mg
Technical specification	≤ 0.05	3.2–4.0	1.5–2.8	0.05–1.0	≤ 0.08	≤ 0.02	≤ 2.0	0.02–0.08
IP7	0.01	3.39	2.32	0.18	0.038	0.008	0.55	0.036
IP8	0.02	3.43	2.25	0.15	0.042	0.009	0.48	0.044
IP9	0.02	3.41	2.41	0.15	0.034	0.005	0.53	0.057
Mean value	0.017	3.41	2.33	0.16	0.038	0.007	0.52	0.046
Standard deviation	0.006	0.02	0.08	0.02	0.004	0.002	0.04	0.011

Table 20: "Recorded maximal deviation of edge distance in" "three PWR inserts" (from Table 5-4 of SKB TR-10-14)"

Tolerance in edge distance – reference design (technical specification)	PWR inserts	Maximum deviation from nominal edge distance (mm)
± 10 (± 5)	IP7	5.5
	IP8	4.0
	IP9	2.9
	Mean value	4.1
	Standard deviation	1.3

Table 21: “Material composition (ppm) of copper ingots for copper tubes manufactured over the period 2005–2008. Contents are specified in ppm apart from the Cu content, which is expressed as a percentage. To the right, the mean value (MV) and standard deviation (STD) have been compiled.” (Table 5-5 in SKB TR-10-14)

Tube no:	Specification	Material composition – large ingot for tubes									
		T45	T46	T47	T48	T53	T56	T57	T58	MV	STD
Man. Year		2005	2005	2005	2005	2007	2007	2008	2008		
Cu	≥99.99	99.99	99.991-99.992	99.99	99.99	99.991–99.992	99.991	99.991	99.992	99.991	0.001
P	30–100	71	67–70	66–72	66–72	60–73	67–72	69–88	54–56	68.4	7.8
O	<5	0.8–1.1	0.7–0.9	0.8–1.2	0.8–1.5	1.0–1.8	0.9–1.3	0.5–0.7	1.6–2.4	1.13	0.49
S	<8	4.8	4.7–4.8	4.5–4.8	4.4	5.3–5.7	4.3	4.3	5.3–5.6	4.77	0.47
H	<0.6	0.3–0.4	0.3–0.4	0.4–0.5	0.3–0.5	0.4–0.6	0.43–0.44	0.28–0.5	<0.1	0.37	0.14
Ag	<25	13	13.6–14.2	13.2–13.4	13.5	13.9-14.1	14.9	14.3-15	13.2	13.8	0.7
As	<5	0.81	0.78–0.81	0.80–0.83	0.82	0.78	0.96–0.97	0.87-0.99	0.85-0.87	0.85	0.07
Bi	<1	0.114–0.116	0.113-0.116	0.109-0.112	0.119-0.120	0.18-0.19	0.20-0.21	0.15-0.21	0.104-0.117	0.14	0.04
Cd	<1	<0.003	<0.003	<0.003	<0.003	<0.003	<0.003	<0.003	<0.003	<0.003	–
Fe	<10	1.4	1.4–1.5	1.4	1.4–1.5	0.6–0.7	0.2–0.4	0.6–0.7	1.1–1.2	1.06	0.44
Mn	<0.5	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	–
Ni	<10	0.7–0.8	0.7–0.8	0.8–0.9	0.8–0.9	0.7–0.8	0.4	0.3–0.5	1.1–1.2	0.74	0.24
Pb	<5	0.24	0.24–0.25	0.27	0.27-0.28	0.32	0.25-0.27	0.18-0.26	0.26-0.29	0.26	0.03
Sb	<4	0.054–0.060	0.053	0.053-0.054	0.06	0.11	0.10-0.11	0.08-0.10	0.06	0.072	0.023
Se	<3	0.2	0.2	<0.09	<0.09	0.3	0.4	0.3	0.1–0.2	0.22	0.11
Sn	<2	0.05-0.06	0.05-0.06	0.05-0.06	0.06-0.07	0.09	0.1	0.06-0.07	0.18-0.19	0.084	0.043
Te	<2	0.05	0.05	0.05	0.05	0.06	0.1	0.07–0.11	0.05	0.063	0.021
Zn	<1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	–

6. Additional specifications in the SKB criticality safety report[1]

The criticality safety report in the SKB application documentation (SKBdoc 1193244) is reviewed separately from the basic technical documents. A purpose is to avoid using conservative or other assumptions in the criticality safety assessment to be mistaken for facts. Editorial mistakes may also have been introduced in SKBdoc 1193244. It is assumed here that the basic technical documents (e.g. TR-10-13, TR-10-14 and TR-10-52) are reviewed by more people than what is the case for SKBdoc 1193244 and that the quality control of specifications meets a higher standard. On the other hand, the conclusions of SKBdoc 1193244 are probably at least as reliable as the conclusions in other documentation that builds on SKBdoc 1193244.

Some of the fuel and reactor operating data in TR-10-13 (the Spent fuel report), e.g. Table A-4 and loading curve information (e.g. Section 6.7), are taken from SKBdoc 1193244. That report also contains references to source documents. They may not be available for this review but the information on the sources are valuable.

Table 22 is taken from Appendix 3 in SKBdoc 1193244, which is assumed to be correct, except for some minor uncertainties (marked in red) and some editorial changes made (also marked in red text or as changes) by the author. Some of the specifications in Table 2 (from TR-10-13) are different to Table 22 specifications and may be more accurate since they are the results of checks. Editorial changes to Table 22 include a change of the fuel type “17x17 HTTP” to “17x17 HTP”and that the “17x17 HTP M5 Monobloc” fuel type is renamed to be the same as in Table A-4 of TR-10-13, i.e. “17x17 HTP X5”. Footnotes 2 and 4 contain minor editorial corrections made by the author.

[1] The text in this chapter is based on, often copied from, SKB source documents

Table 22: PWR fuel type specifications (basis for Table 2) (from Appendix 3 of SKBdoc 1193244).

Fuel type	W15x15	KWU15x15	F15x15 AFA3G	15x15 AGORA	W17x17	AA17x17	F17x17	S17x17 HTP	17x17 HTP	17x17 HTP M5	17x17 HTP X5	17x17 AFA3G
Table footnote reference	1	1	2	3	1	1	9	4	5	6	7	10
No of fuel rods	204	204	204	204	264	264	264	264	264	264	264	264
Fuel rod pitch (mm)	14.3	14.3	14.3	14.3	12.6	12.6	12.6	12.6	12.6	12.6	12.6	12.6
Fuel rod outer diameter (mm)	10.72	10.75	10.72	10.77	9.5	9.5	9.5	9.55	9.55	9.5	9.5	9.5
Fuel rod inner diameter (mm)	9.48	9.3	9.484	9.505	8.36	8.36	8.36	8.33	8.33	8.36	8.36	8.36
Cladding thickness (mm)	0.62	0.725	0.618	0.6325	0.57	0.57	0.57	0.61	0.61	0.57	0.57	0.57
Pellet diameter (mm)	9.29	9.11	9.294	9.33	8.19	8.19	8.19	8.17	8.165	8.192	8.192	8.192
Cladding material	Zr4	Zr4	M5	Zr4	Zr4	Zr4	Zr4	Zr4	Zr4	M5	M5	Zr4
Active fuel length (mm)	3658	3658	3658	3658	3658	3658	3658	3658	3658	3658	3658	3658
UO2 density (g/cc)*	10.22	10.46	10.52	10.52	10.45	10.45	10.45	10.45	10.45	10.52	10.55	10.52
No of guide tubes	20	20	20	20	24	24	24	24	24	24	24	24
Guide tube material	Zr4	Zr4	M5	M5	Zr4	Zr4	Zr4	PCAm	PCAm	PCAm	PCAm	Zr4
Guide tube outer diameter (mm)	13.87	13.86	14.1	14.1	12.24	12.09	12.05	12.24	12.24	12.24	12.45	12.45
Guide tube inner diameter (mm)	13.01	13	13.05	13.05	11.44	11.18	11.25	11.3	11.3	11.3	11.45	11.45
Guide tube cladding thickness (mm)	0.43	0.43	0.525	0.525	0.4	0.455	0.4	0.47	0.47	0.47	0.05	0.5
No of instrument tubes	1	1	1		1	1	1	1	1	1	1	1
Instrument tube material	Zr4	Zr4	M5	M5	Zr4	Zr4	Zr4	PCAm	PCAm	PCAm	PCAm	Zr4
Instrument tube outer diameter (mm)	13.87	13.86	14.1	14.1	12.24	12.09	12.05	12.24	12.24	12.24	12.45	12.45
Instrument tube inner diameter (mm)	13.01	13.03	13.05	13.05	11.428	11.428	11.25	11.3	11.3	11.3	11.45	11.45
Instrument tube cladding thickness (mm)	0.43	0.43	0.525	0.525	0.406	0.406	0.4	0.47	0.47	0.47	0.5	0.5

1. CLAB 96 - Dataunderlag för kriticitetsberäkningar, Agrenius Ingenjörsbyrå AB, augusti 1991
2. Areva FF DC 02916 Transport and reprocessing document for reload SSPK of reload SSPK of Ringhals 2 fuel assemblies 15x15AFA3GAA

3. Areva A1C-1332397-0 NP fuel assemblies delivered to Ringhals 2/31/07
4. Fuel type data for final storage - PWR - Siemens HTP Ringhals 3 2000-06-16
5. Areva A1C-1313665-4 Reprocessing information for Framatom ANP fuel assemblies delivered to delivered to Ringhals 3/4
6. Areva A1C-1333871-0 NP fuel assemblies delivered to RH 3/25/08
7. Areva A1C-133864-0 NP fuel assemblies delivered to RH 3/24/07
8. Fuel type data for final storage - PWR - reload 18 / SUPW Ringhals 4
9. ABB BR 91-446 Criticality calculations: PWR Compact canisters (Clab 96), 1991-10-28
10. Fuel Type Data for Final Storage - PWR - Reload 18 / SUPW Ringhals 4 17x17AFA3

SKBdoc 1193244 refers to a reference /7/ (ORNL/TM – 1999/99) where the results show that the predicted k_{eff} value increases if lower specific power is used. SKB used “a relatively low power density of 14 MW/assembly” (slightly different to the 15 MW/assembly specified in Table 7 of the same report and shown in Table 23 below). This is compared to the Ringhals 2 value which in average is 17 MW/assembly and to the Ringhals 3/4 value 18 MW/assembly which will be increased to 20 MW/assembly.

Some PWR-fuel assemblies contain integral burnable poison. The SKBdoc 1193244 refers to another reference /8/ (NUREG/CR-6760) that shows that k_{eff} for fuel containing Gd_2O_3 is always lower than the multiplication factor for fuel without Gd_2O_3 throughout burnup. Burnable poison was thus not modelled in the SKB criticality safety assessment.

In order to calculate the isotopic composition of the fuel at different burnup the fuel had to be subjected to different burnup histories. The main parameters for the depletion calculations are shown in table 23.

The burnup of a fuel assembly is always the assembly average burnup if nothing else is stated.

Table 23: “Main parameters for the depletion calculation” (from Table 7 of SKBdoc 1193244)

Parameter	PWR
Assembly power (MW)	15
Avg. fuel temperature (°C)	625
Coolant pressure (bar)	155
Coolant temperature (°C)	304
Boron concentration (ppm)	600
Coolant density (kg/dm3)	0.68
Cycle length (days)	345
Shutdown length (days)	20
Decay time (yrs)	1

(Sources: Ringhals 2007-10-19, 1960160/1.1 and OKG 2008-05-26, reg nr 2008-14670. Confidential information. Available only for the Swedish Radiation Safety Authority.)

In initial PWR-cores in Ringhals burnable poison rods were used in about 60 of the 157 fuel assemblies. The poison rods are made of stainless steel, borosilicate glass and zircaloy. The SKBdoc 1193244 refers to yet another reference /9/ (NUREG/CR-6761) that shows that the presence of burnable poison rods gives a higher k_{eff} compared with fuel without poison rods throughout burnup (after removal of those rods). The burnable poison is depleted during the first cycle. If the burnable rod cluster is not removed after the first cycle a significant portion of the reactivity difference is shown to be due to the displacement of moderator. The reactivity difference is shown to be up to 3% Δk. This has to be considered when fuel assemblies that have contained burnable poison rods will be compared with the loading curve.

In addition to the above given information, the following quotes are selected from SKBdoc1193244 for being of major interest:

“Normally during operation control rods in both BWR and PWR are not inserted in the core. The effect of inserted control rods has therefore not been evaluated.” (Section 9.9 of SKBdoc1193244)

“The declared assembly average burnup is based on the plant heat balance, measurements and calculations of the power distribution in the core. Based on uncertainties of the measurements and calculations the uncertainty in the burnup prediction is estimated to be within σ_{BU}=2% for BWR and $2\sigma_{BU}$=3.65% for PWR. (Sources: OKG 2008-05-26, reg nr 2008-14670 and Ringhals 2007-10-19, 1960160/1.1. Confidential information. Available only for the Swedish Radiation Safety Authority.)” (Section 9.5 of SKBdoc1193244)

“Due to the higher temperature and lower moderator density in the top of the core more Pu-239 will be produced than in average” (Section 9.6). In SKBdoc 1193244 (Section 9.6) the core exit temperature and the corresponding water density was used (for PWR).

The “axial burnup distributions from 15 cores from Ringhals 2, 3 and 4 were studied. In addition 9 cores from the Great- and Frej-projects were studied, see appendix 4. The fuel types are 15x15 and 17x17-fuel with burnup from 10 MWd/kgU up to 65 MWd/kgU. Initial enrichments are 3.2 – 4.95% U-235.

From this population a number of distributions were chosen for analysis. Distributions with the highest and lowest peaking factors (F), with the lowest burnup in the bottom node, with the lowest burnup in the top node were selected. A bounding burnup distribution was constructed by reducing the burnup in the bottom and top node by 20% while keeping the assembly burnup constant. The resulting distributions are shown in figure” 19 (Section 9.8 of SKBdoc1193244).

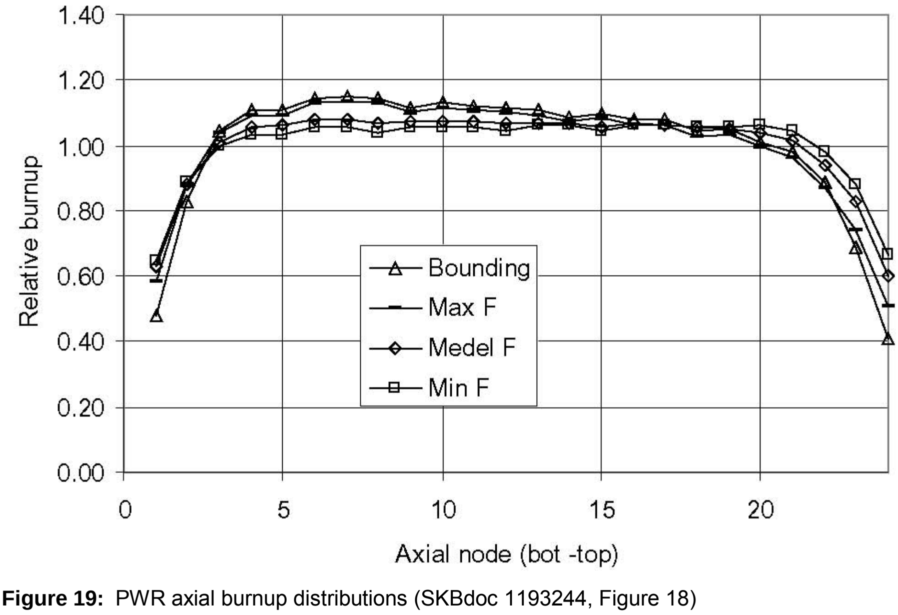


Figure 19: PWR axial burnup distributions (SKBdoc 1193244, Figure 18)

“It should be noted that the radial difference in the burnup from the average is ± 10% in the calculations which is higher than values reported in sources: Ringhals

2007-10-19, 1960160/1.1 and OKG 2008-05-26, reg nr 2008-14670. Confidential information. Available only for the Swedish Radiation Safety Authority." (end of Section 9.10 in SKBdoc 1193244)

Section 5.2 of SKBdoc 1193244 contains material data for the canister insert materials cast iron (SS 140717), the square steel tubes (S355J2H) and the steel lid (S355J2). This information is provided here in Tables 24-26.The specifications for the steel tubes have not been found in any other documentation.

Table 24: SS 140717 (Table 2 in SKBdoc 1193244)

Material	C	Si	Mn	P	S	Ni	Mg
Min (%)	3.2	1.5	0.05	-	-	0	0.02
Max (%)	4	2.8	1	0.08	0.02	2	0.08

Fe-content 90.02 -95.13% (balance)

Density 7.1 g/cm^3

Table 25: S355J2H (Table 3 in SKBdoc 1193244)

Material	C	Si	Mn	P	S
Max (%)	0.22	0.55	1.6	0.03	0.03

Fe-content 97.57% (balance)

Density 7.85 g/cm^3

Table 26: S355J2 (Table 4 in SKBdoc 1193244)

Material	C	Si	Mn	P	S	Cu
Max (%)	0.24	0.6	1.7	0.035	0.035	0.6

Fe-content 96.79% (balance)

Density 7.85 g/cm^3

"The disposal canister shell is made of pure copper, density 8.9 g/cm^3." (Section 5.2 in SKBdoc 1193244)

Section 6.3.1 of SKBdoc 1193244 covers a nodular cast iron composition of >90 wt.% iron, <6 wt.% carbon and <4 wt.% silicon. The conclusion in TR-10-13 (the Spent fuel safety report) has been changed to <4.5 wt.% carbon and <6 wt.% silicon.

7. SKB calculation methods[1]

Version 5.1 of the SCALE calculation system (same report number but older edition than reference 8 in this Technical Note) was used by SKB. For depletion calculations (determining the nuclear properties of the fuel after use in a reactor), the sequence SAS2 was used. For k_{eff} calculations the code STARBUCS and the sequence CSAS25 were used together with the 44-group ENDF/B-V library. All codes and data are included in SCALE 5.1.

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[1] The text in this chapter is based on, often copied from, SKB source documents

The methods used by SKB are developed with criticality safety as the main application. They were developed at Oak Ridge National Laboratory (ORNL) with financial support mainly from Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) in U.S.A. There are many publications on the development, validation and use of SCALE. The specific sequences, codes and data used by SKB have been used by ORNL and NRC as well as by many license applicants in the U.S.A.

The SKB validation of SCALE 5.1 for application to criticality safety evaluation of the SKB final disposal canister for PWR fuel involved calculations of 59 benchmarks based on critical experiments. The source of the calculation input data for the benchmarks is not documented in SKBdoc 1193244. The bias estimated by SKB is -70 pcm while the uncertainty (one sided upper 95/95 % tolerance level or almost exactly 2σ) is 930 pcm. The corresponding uncertainty due to Monte Carlo simulation is estimated to be 90 pcm.

The SAS2 sequence is old and simple but was considered adequate for the safety analysis of the Yucca Mountain Project.

SAS2 has also been used by participants in studies by the OECD/NEA expert group on burnup credit (reference 9. Even though there have been significant errors for individual nuclide densities, the overall k_{eff} values have been quite good, considering the many approximations. EMS has participated in the OECD/NEA expert group since the start but not with depletion calculations.

An EMS reason for not contributing with SCALE calculation results for previous OECD/NEA depletion studies are that the SCALE code package has been used by other participants. The method has been under development and not very easy to use for BWR fuel. Another reason is that there has not been any urgent need for burnup credit reviews in Sweden.

SKB has to some extent validated the use of burnup credit against proprietary reactor measurements. The two reports related to estimated burnup uncertainties have not been available for review. The burnup prediction is estimated to be within 2% (1σ or one standard deviation) for BWR and 3.65% (2σ or two standard deviations) for PWR. SKB has estimated that the reactivity (change of k_{eff}) varies linearly with the burnup both for PWR and BWR fuels. At 50 MWd/kgU the 2σ reactivity uncertainties are estimated to be 105 pcm (0.00105) for PWR fuel and 131 pcm (0.00131) for BWR fuel, both with uranium containing 5 wt.% ^{235}U before use in the reactor. The uncertainties increase slightly with lower enrichment.

SKB has also used ORNL studies of radiochemical measurements of individual actinide and fission product nuclide assays in used fuel samples to derive correction factors and uncertainty estimates. An SKB summary of the results, provided in Section 9.14 of SKBdoc 1193244, is that when adding additional actinide nuclides only, there is a constant k_{eff} error (compensated for by a correction factor) from zero burnup to 50 MWd/kgU of about 300 pcm. When both additional actinide and fission product nuclides are added, an approximate linear k_{eff} error starts with 300 pcm at zero burnup and reaches about 900 pcm at 50 MWd/kgU.

The validation technique used by SKB and developed by ORNL/NRC does not appear to be sufficient neither for unused fuel nor for used fuel. For the unused fuel, the major concern is correlation between error sources of many of the benchmarks. For used fuel, the lack of benchmarks based on real fuel in critical configurations is apparent. Some of the concerns have been expressed by the author in a previous SSM Technical Note 2012:65 for GLS (reference 10) and in a review report for SSM on Clink (Mennerdahl, August 2012, reference 11).

The purpose of this study is not to discuss those issues but to make as independent calculations as reasonable of the intact, water-filled PWR copper canister with unused and used PWR fuel.

8. SKB calculations for the PWR canister[1]

The following paragraph is a quote from TR-10-13 (the spent fuel report, p. 32):

"In those calculations, combinations of enrichments and burnups that will result in a multiplication factor (k_{eff}) of 0.95 for the encapsulated assemblies are derived /SKBdoc 1193244/. The calculations are based on typical PWR assemblies, i.e. Areva 17×17, and on the reference design of the canister and typical properties of bentonite and rock. Further, it is assumed that identical assemblies occupy all positions in the canister. In the calculations, a systematic investigation of uncertainties is made. All parameters with potential impact on the criticality are investigated. Parameters that can be shown to be insignificant are set to a typical value while parameters that are significant are set so as they favour criticality."

In the quoted text above, the k_{eff} value refers to encapsulated assemblies (canisters with assemblies) and not to individual assemblies.

The calculated k_{eff} value for a water-filled canister with PWR fuel and reflected by water is 1.0872 (section 6.4.1. of SKBdoc 1193244).

The calculated k_{eff} value for a water-filled canister with PWR fuel and reflected by bentonite is 1.0888 (section 6.4.4. of SKBdoc 1193244).

A large number of perturbation results are covered. One of them is a reduction of the compartment size from a 23.5 cm square to a 22.99 cm square (section 6.6.2. of SKBdoc 1193244). For the bentonite-reflected canister the k_{eff} value increased by 0.0044.

A design case was determined as an F15x15AFA3G fuel assembly, with significant parameters (e.g. temperature at 277 K) at their bounding conditions while other parameters were at nominal values. Section 6.8 of SKBdoc 1193244 specifies the list of values for about 18 different parameters. The selection of parameter values is based on calculations.

For the design canister the k_{eff} value is 1.1041 (section 6.8 of SKBdoc 1193244). The design case is conservative, without necessarily being unrealistic, and the increase in k_{eff} from about 1.087 to about 1.104 is typical for calculations of this type (author's comment).

In burnup credit assessment, the selection of nuclides included (section 7 of SKBdoc 1193244):

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[1] The text in this chapter is based on, often copied from, SKB source documents

U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Np-237 Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Eu-151, Sm-152, Eu-153 and Gd-155

The decay time was one year after the last cycle.

Figure 20 shows the influence of depletion on k_{eff}.

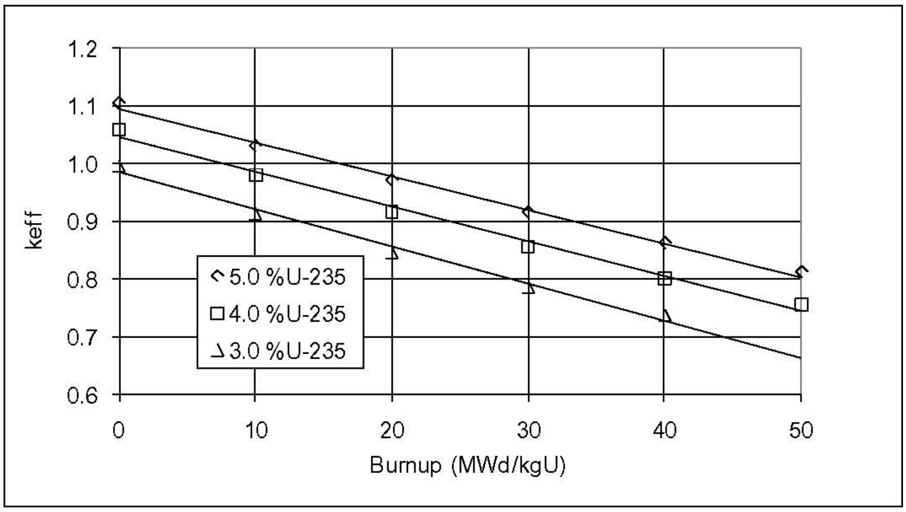

Figure 20: k_{eff} as a function of burnup, actinides and fission products PWR (SKBdoc 1193244, Figure 16)

Figure 21 shows the influence on k_{eff} of using a uniform burnup distribution versus using a bounding distribution. The results refer to cases with selected actinide nuclides and fission products. Figure 19 above shows the meaning of Max F and bounding.

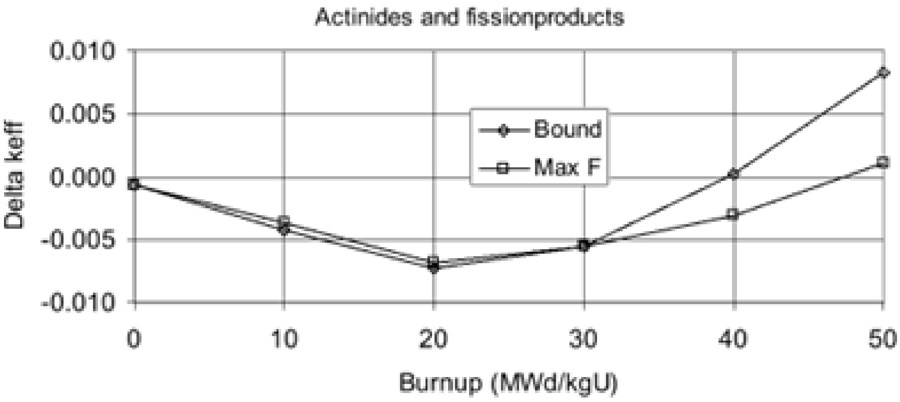

Figure 21: The end effect as function of burnup for different axial burnup distribution for PWR (SKBdoc 1193244, Figure 20)

Figure 22 shows the influence of long decay times on k_{eff}. Figure 23 shows the sensitivity to the axial burnup distribution.

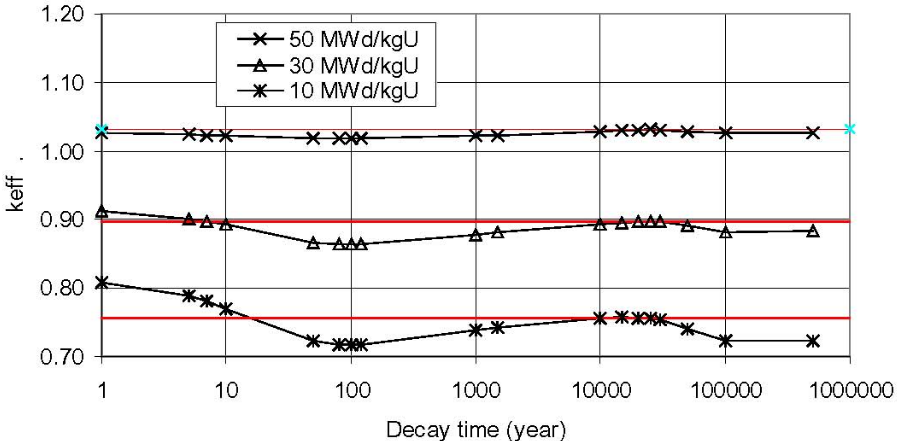

Figure 22: PWR long term reactivity change for different burnup, actinides and fission products (SKBdoc 1193244, Figure 35)

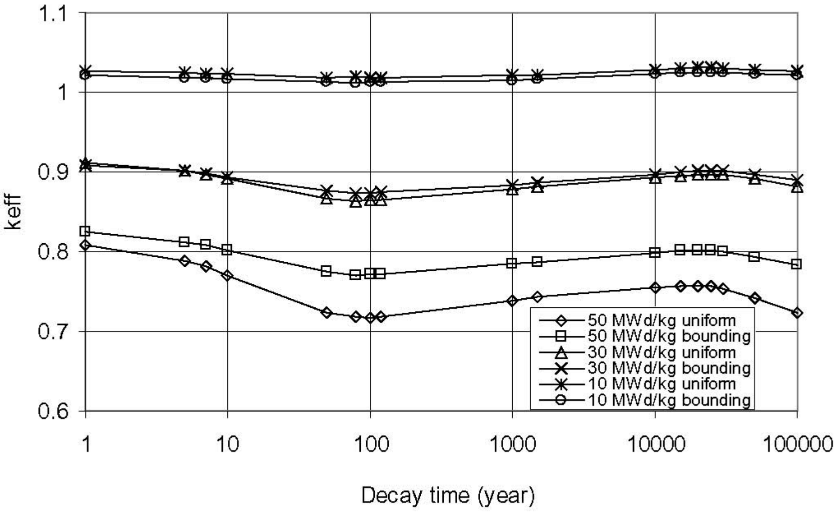

Figure 23: PWR k_{eff} vs. decay time for bounding and uniform axial burnup distribution, actinides and fission products (SKBdoc 1193244, Figure 38)

9. EMS calculation methods

9.1. General description of methods

The calculation methods used by EMS are from two different sources. They are both from the U.S.A. but have different backgrounds and are essentially independent. This is important for a safety reviewer since the capability to use independent methods from the applicant is valuable. Either or both methods may be used as needed or found to be informative. In this review only SCALE 6.1.1 was used.

The SCALE 6.1.1 (reference 8) calculation system is the primary method. SCALE includes many computer codes, data libraries and control sequences. The current version and previous versions have been used since the first release of SCALE 0 around 1980. Even a few years before that, the author used many of the codes and cross-sections to become available in SCALE 0. The development of SCALE has primarily been focused on criticality safety but also shielding and other areas (previously also heat transport) are covered. SCALE 6.1.1 contains several options for fuel depletion calculations and for determination of k_{eff}.

The other method is based on a single computer code, MCNP5 release 1.60 (reference 12). MCNP has been used by EMS for about ten years but has a much longer history. It is a more rigorous Monte Carlo code than the ones included in SCALE. It is used in many areas, including high-energy neutron transport (fusion, accelerators), shielding, medical evaluations, etc. MCNP5 is now being merged with the more general MCNPX version and a first formal release of the merged code is expected in 2013 (a beta version is publicly available).

A major advantage of using MCNP5 is that there are many continuous energy cross-section libraries available. They include various ENDF/B libraries (U.S.A.), JEF libraries (Europe) and JENDL libraries (Japan). Even some CENDL (China) cross-sections have been used by EMS since they appear to have the best copper cross-sections.

MCNPX includes some capability for fuel depletion calculations but it is too early to say if and how MCNP6 may be applied for burnup credit. Currently, MCNP5 is often used internationally together with independent depletion codes, primarily ORIGEN-2 which is an old version of the ORIGEN version in SCALE. EMS has not used such coupling between MCNP and ORIGEN. Instead another combination (not yet released) has been selected for EMS use in 2013. This combination will include a depletion method that is independent of ORNL methods.

MCNP5 has not been used in the current review and is not expected to give any different results if the same cross-section source is used (e.g. ENDF/B-VII.0).

9.2. Validation standards for calculation methods

ANSI/ANS 8.24 (validation of criticality safety calculation methods, reference 13) and ANSI/ANS 8.27 (burnup credit, reference 14) are the primary standards selected by the author. There is also an ISO standard for PWR burnup credit that is considered (reference 15).

The depletion and k_{eff} calculations are preferably made in one single sequence when the method supports this. This simplifies the bias and uncertainty propagation as shown below. If not possible, a second choice is to use the same cross-section library both for depletion and for k_{eff} calculations.

A traditional subcriticality requirement can be shortly expressed as:

$$k_p + \Delta k_p \leq k_c - \Delta k_c - \Delta k_m \qquad (1)$$

The terms in the equation are:

k_p The estimated value of k_{eff}, based on calculation and, when applicable, including an allowance for non-conservative modelling assumptions.

Δk_p An allowance for calculation uncertainty.

k_c The calculated value of k_{eff} for a set of benchmarks, normalized to a benchmark set value (usually 1.000 or close to 1.000). Includes biases (not corrections for biases) .

Δk_c An allowance for the uncertainty in k_c. Includes benchmark uncertainties and benchmark calculation uncertainties.

Δk_m An allowance for additional uncertainties, often from unknown error sources.

ANS 8.27 has an expanded subcriticality requirement that is expressed as:

$$k_p + \Delta k_p + \Delta k_i + \Delta k_b \leq k_c - \Delta k_c - \Delta k_x - \Delta k_m \quad (2)$$

with the additional terms:

Δk_i An uncertainty allowance for nuclide inventory density uncertainties.

Δk_b An allowance for burnup uncertainty.

Δk_x An allowance for additional biases and uncertainties due to nuclide inventory cross-sections that are not included in k_c or Δk_c.

The bias and uncertainties in the combined approach do not rely on nuclide inventory densities and cross-sections. Any additional uncertainties due to depletion may be included directly in k_c and Δk_c or be accounted for separately in the term Δk_d:

Δk_d An allowance for additional biases and uncertainties due to burnup credit calculations that are not included in k_c or Δk_c. This term replaces Δk_i or Δk_x when a combined depletion and k_{eff} validation is made.

The term Δk_d is added to account for additional biases and uncertainties when justified. The equation (1) now reads:

$$k_p + \Delta k_p + \Delta k_b \leq k_c - \Delta k_c - \Delta k_d - \Delta k_m \quad (3)$$

The validation method preferred by EMS is to rely on a combined approach for depletion and k_{eff} determination using combined benchmarks and a combined calculation method. Verification of individual nuclide concentrations and nuclide cross-sections changed or introduced by depletion, can be made against radiochemical assay data benchmarks. That is considered to be a valuable information source, in particular to cover accidents, long-term storage and final disposal, but is in itself not judged by the author to be sufficient for validation.

The combined validation method is considered to be the traditional method used for validation of calculation methods for specific criticality safety applications. The k_{eff} value of a total system is of primary interest, not the reactivities of detailed components. The ANSI/ANS 8.24 (validation) and 8.27 (burnup credit) standards support the combined approach.

The total macroscopic cross sections of appropriately homogenized regions of the fuel are needed, not detailed cross sections and inventories for hundreds of nuclides in each such region. The depletion is a significant complication but is not necessarily a large error contributor.

Flux-weighted homogenization of fuel lattices is a traditional method and it doesn't provide accurate individual, nuclide cross-sections and nuclide inventories but a sufficiently accurate combination.

9.3. Validation of k_{eff} calculation methods

There are many application areas for SCALE 6.1.1 and MCNP5 and they are validated by EMS in accordance with the need for such validation. Each new version needs to be validated but often previous validation applies and this only needs to be confirmed.

The ICSBEP Handbook (OECD/NEA September 2011, reference 16) is the primary source for benchmarks. The IRPhE Handbook (OECD/NEA March 2012, reference 17) will become more important in the future. International studies (e.g. OECD/NEA) and standards development (ANS, ISO and IAEA) provide opportunities for comparing and testing the actual use of the method. Since each method can be used in different ways, the validation must account for each specific user of the method.

9.4. Validation of fuel depletion methods

Nuclear reactors that generate energy also deplete (or rather transmute) the fuel. The nuclide inventory is changed and, at least locally, a uniform or homogeneous distribution will change into a geometrically continuously varying composition. The term "depletion" is usually used by reactor physicists when they refer to the transmutation of the fuel in the reactor. After the reactor operation there will be radioactive decay of the radioactive nuclides.

Prediction of the depletion of the reactor fuel has been a necessary task for reactor physicists. The capability of making such predictions have been developed and routinely tested during 70 years (the first criticality of a man-made nuclear reactor was in 1942). Calculation methods (e.g. CASMO and PHOENIX) developed in Sweden (Atomenergi, Studsvik, Asea-Atom, etc.) have been very successful in the design of fuel and use at many nuclear power plants all over the world. CASMO and PHOENIX have also been used successfully in criticality safety analysis of fuel storage pools and for transport package designs.

Validation of depletion methods are normally made by the vendors of the methods and by the end users of the methods. The benchmarks are often actual operations and measurements, not experiments, and they include prediction of k_{eff} and of many other variables and parameters. Unfortunately, almost all measurement specifications are proprietary to their owners.

Some measurements involving power reactor operation have been published. The information is so complicated that it has been difficult to apply using popular criticality safety methods. Even so, this approach appears to be the best for burnup credit application. Recent developments by EPRI (reference 19) and TVO (reference 18) have demonstrated that the complicated reactor measurements can be simplified considerably for validation.

Another validation approach is to take samples from the used fuel and make radiochemical analysis to determine the assays of some nuclides in the samples. This is the approach selected by ORNL, with strong support from NRC (reference 20), and also the approach considered by the OECD/NEA expert group on burnup credit. In a way it is understandable since such data are publicly available and it seems to be a scientific approach (important for the OECD/NEA expert group).

The validation approach based on radiochemical assay analysis contains many large uncertainties. It is not a direct validation approach, rather an effort to verify all significant method components and to obtain some kind of validation from that. It is not the traditional criticality safety approach, it is perhaps not sufficient and it has certainly not been necessary in order to apply burnup credit safely to short-term storage and transport.

The major reasons for the slow development of burnup credit methods and applications appear to be lack of industry interest combined with the criticality safety community focus on publicly available methods and benchmarks. Burnup credit is primarily an economical issue but is also of importance for preservation of natural resources.

EMS has focused on the direct validation approach, using power reactor measurements and comparison with methods that have been validated against such measurements. This is the traditional method used for validation of methods used for burnup credit and burnable absorber credit both in Sweden (e.g. Clab since 1995) and in other countries. It is also supported by ANSI/ANS 8.24 and 8.27 standards. The other approach is valuable as a complement and becomes more important when accidents and final disposal are considered. Studies of the stability of the fuel nuclide inventory as a function of normal handling, events and time are required for both approaches.

9.5. Validation of SCALE 6.1.1 k_{eff} calculations

Validation of SCALE 6.1.1 codes like KENO V.a (Monte Carlo), KENO-VI (Monte Carlo), NEWT (2D deterministic), XSDRNPM/S (1D deterministic) relies on adequate cross-sections and representative benchmarks. In general, the biases in k_{eff} are less than 0.01 (1000 pcm) and often much lower than that for the most recent ENDF/B-VII.0 cross-section library.

For this project, validation of PWR fresh (unused) fuel assemblies in water is essential. Presence of stainless steel, nodular iron and copper also should be considered.

Recently evaluated ICSBEP Handbook benchmarks (e.g. those based on the Studsvik FR0 experiments) where copper has a significant effect on k_{eff} as a reflector have been presented by the author at ICNC 2011 (reference 21). They show that the copper cross-sections appear to have large uncertainties. However, for the intact canisters, the copper appears to have a very small influence on k_{eff}.

Calculations of benchmarks including light water reactor fuel rods with water and/or iron as reflectors have been made previously without any indication of dramatic uncertainties or biases.

Results of validation against ICSBEP Handbook benchmarks are not presented in this report.

The influence of changes of the fuel due to reactor use (depletion) is validated separately. Validation of MCNP5 (Monte Carlo) is not covered in this Technical Note.

9.6. Validation of SCALE 6.1.1 depletion calculations

Validation of the SCALE 6.1.1 main depletion control module TRITON with one or more of the calculation sequences T-DEPL (based on NEWT), T5-DEPL (based on KENO V.a) and T6-DEPL (based on KENO-VI) is covered here. The separate sequence STARBUCS based on simplified burnup credit analysis has been used in this review but no specific validation has been made.

T-DEPL is the most advanced sequence. The 2D deterministic calculations with NEWT require some approximations that need to be tested with 3D Monte Carlo calculations (KENO-VI in this review). A reason for using KENO-VI rather than KENO V.a is that KENO-VI and NEWT use the same geometry format and that KENO-VI has more advanced geometry capabilities. It is expected the KENO V.a and KENO-VI give essentially identical results. KENO-VI is considerably slower than KENO V.a.

9.6.1. OECD/NEA calculation benchmarks

The OECD/NEA/NSC/WPNCS expert group on burnup credit has since 1991 made several depletion studies (reference 9). They are referred to in the order of time as:

- Phase I-B (1D PWR fuel rod cell);
- Phase III-B (BWR 2D 8x8 fuel lattice with gadolinium);
- Phase II-D (2D PWR lattice with absorber rods);
- Phase IV-B (MOX fuel in 1D fuel rod cells and in 2D lattice super-cells with one MOX lattice and three UO_2 lattices) and
- Phase III-C (BWR 2D 9x9 fuel lattice with gadolinium).

The studies refer to benchmarks but they are not based on direct experiments. They are developed using established methods and based on realistic fuel and reactor operating conditions. Some previously validated methods were used by some participants. It is often difficult to determine a best-estimate result for most of these benchmarks but the range of results is sufficiently limited to be valuable for validation of SCALE 6.1.1.

The main purpose of this renewed interest in the OECD/NEA calculation exercises is to test different options in SCALE 6.1.1 and to assure that they are used properly. The need for accuracy is not high at this time. The author has not previously made calculations for the OECD/NEA depletion benchmarks.

The OECD/NEA benchmarks are briefly presented in Appendix A.

Concerning the influence of long radioactive decay times, another study by the OECD/NEA /NSC/WPNCS expert group on burnup credit compared the influence of decay times up to one million years concerning k_{eff} and nuclide inventories:

- Phase VII (final disposal considerations of burnup credit),

The results have been published by ORNL as an OECD/NEA report. EMS contributed to this study, supported by SSM (SKI at that time). There were no dramatic uncertainties revealed by that study.

9.6.2. EPRI and IRPhEP benchmarks

The main source of validation benchmarks used by EMS for burnup credit and burnable absorber credit is the large EPRI study made by Studsvik and published in 2011 together with a related EPRI evaluation for the IRPhE Handbook (a draft evaluation will be published in the spring 2013 edition of the handbook).

The EPRI and IRPhE benchmarks are briefly presented in Appendix B. Since the IRPhE benchmarks are still under development, the details are not provided in this Technical Note. The IRPhE benchmarks from EPRI are more strictly based on measurements than the published EPRI benchmarks. The EPRI benchmarks are more closely representative of burnup credit applications.

9.6.3. Recent NRC SFST ISG-8 Rev.3 validation approach

NRC SFST ISG-8 Rev.3 uses radiochemical assay data measurements in an effort to demonstrate validation of burnup credit methods. The Guide requires access to the proprietary French HTC benchmark experiments to fully apply the NRC/ORNL validation efforts for use with SCALE 6.1. Those experiments are available to U.S.A. organizations after approval by ORNL. In other countries it appears as if a licensing agreement with the French IRSN is required. The NRC/NUREG efforts are valuable as complements to industry reactor-based benchmarks based on measurements. This is particularly true for scenarios where the fuel is no longer intact.

9.6.4. Summary of EMS validation of burnup credit methods

A preliminary summary of the calculations for burnup values higher than 15 MWd/kg is that k_{eff} is determined within 0.01 of the average OECD/NEA benchmark results and within 0.005 of the EPRI and IRPhE benchmark results.

For BWR fuel with burnup values near the peak reactivity around 10 MWd/kg, the differences are larger, up to 0.02, with the new SCALE 6.1.1 results being higher. There are currently no easily applicable benchmarks based on experiments publicly available for BWR depletion calculation validation. Some proprietary benchmarks (in particular for cold reactor conditions) for BWR may be valuable for PWR fuel burnup credit validation.

10. EMS review calculations

10.1.1. Purpose

The purpose of the calculations is to evaluate the neutron multiplication factor (k_{eff}) for the copper canister with PWR fuel. The canister material and geometry with fuel are all intact. However, the inside is assumed to be flooded with water which is a design-basis condition. The fuel may be of different designs and may be unirradiated (fresh) or depleted to different burnup values.

10.1.2. Canister design

The basic canister design appears to be very simple for criticality safety assessment. The canister, excluding the fuel, contains the copper shell and a cylindrical insert with a steel lid. This insert is created from a four square steel tubes (temporarily sand-filled) in a cassette that is flooded with nodular cast iron and closed. After removing the sand, the four channels can each hold a PWR fuel assembly.

The preliminary geometry data used for the review calculations are presented in Table 27. The material data are presented in Table 28. The geometry model, as shown in Figure 24, is simplified without causing any significant difference in k_{eff} values. The limiting data were used for the materials.

Table 27: Dimensions at initial state

Component	Material	Geometry shape	Parameter	Dimension[1] (cm)
Canister shell – No flanges at the lid and the base	Copper	Cylinder shell (tube, lid and base without flanges)	Outer diameter	105.0
			Inner diameter	95.2
			Outer height	467.5 (+454.5/-13.0)
			Inner height	457.5 (+449.5/-8.0)
Insert	Nodular iron	Solid cylinder with tube and base and four channels for cassette	Diameter	94.9
			Outer height	457.3 (+449.3/-8.0)
			Inner height (without lid)	449.3 (+449.3/0)
	Stainless steel	Cassette with four boxes in symmetrical positions	Box outer sides	25.1 x 25.1
			Box inner sides	22.6 x 22.6
			Box height	444.3 (+443.0/0)
			Box C-C distance	37.0 (before casting)
	Stainless steel	Cylinder steel lid	Diameter	91.0
			Thickness	5.0 (+449.3/444.3)
Fuel	UO2	17x17 lattice	See Table 22	See Table 22
	Zircaloy			

[1] A coordinate system with the origin in the centre of the cassette lower end is selected.

Table 28: Canister materials

Material	Component	Density (g/cm^3)	Elements	Weight fractions (wt.%)
Copper	Canister shell	8.9	Cu	100.0
Nodular iron (limiting values in parenthesis)	Insert	7.1	Fe	93.07 (90.0)
			C	4.0 (4.5)
			Si	2.8 (5.5)
			Mg	0.08 (0)
			Mn	0.05 (0)
Stainless steel	Square tube and lid (lid fractions slightly different, in parenthesis)	7.85	Fe	97.57 (96.79)
			C	0.22 (0.24)
			Si	0.55 (0.6)
			Mn	1.6 (1.7)
			P	0.03 (0.035)
			O	0.03 (0.035)
			Cu	0 (0.6)

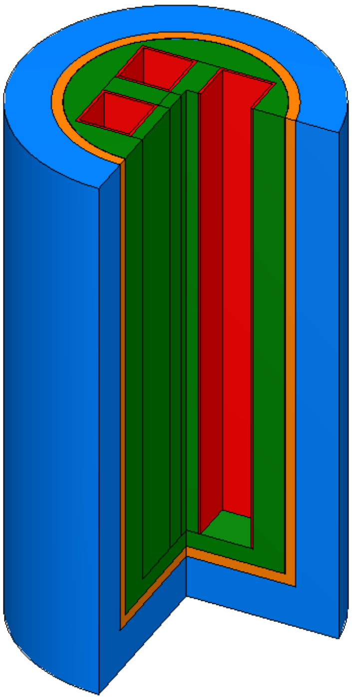

Figure 24: KENO V.a PWR fuel canister (top half and front right 1/4 plus a bit more cut away) without fuel

10.1.3. PWR fuel

The safety documents, e.g. the source for Table 22 of this Technical Note, contain some fuel variations, even for the same type of fuel. The review calculations are based on 17x17 fuel assemblies similar to Areva 17x17 and Westinghouse 17x17 fuel assemblies. The details of the upper and lower ends, outside the fuel regions of the assemblies, are not described in the safety documentation. Those fuel assemblies are very similar to the PWR fuel used in the EPRI and IRPhEP benchmarks. The Westinghouse 17x17 fuel assembly is included in the SCALE library used by STARBUCS.

The review calculations used a simple approximation of the PWR fuel assembly, see Figures 25 and 26. The fuel region length was set to 365.8 cm and the total assembly length to 400 cm. The 17.1 cm end regions contained the same lattice but with the fuel and gap replaced with zircaloy 4.

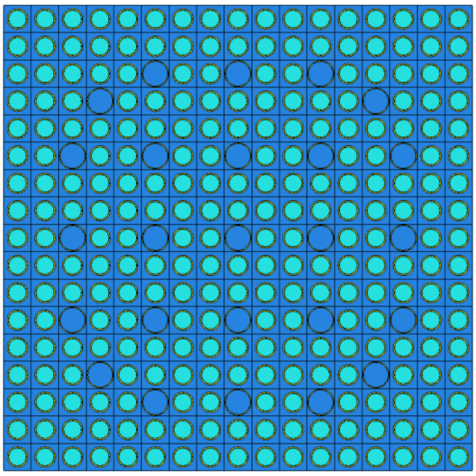

Figure 25: KENO V.a PWR fuel assembly lattice

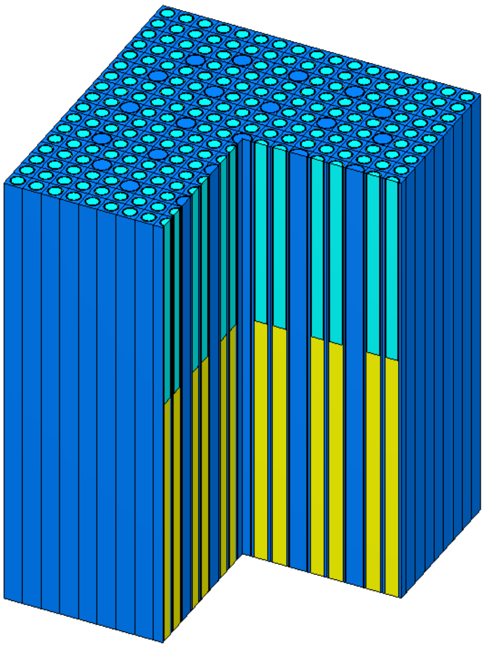

Figure 26: KENO V.a PWR fuel assembly lower end model (front right quarter cut away)

10.1.4. PWR depletion in Ringhals 2, 3 and 4

The general safety documents contain only limited specifications for the reactor depletion of the fuel. SKBdoc 1193244 contains more detailed information and that was applied in the review calculations. The reactor lattice geometry has not been found and that is an important input. It is assumed that this lattice is very similar to that in the reactors that the EPRI benchmarks are based on (McGuire 1 and 2, Catawba 1 and 2). Table 29 contains the basic depletion parameters used in the EMS depletion calculations.

Table 29: Main parameters for the EMS depletion calculations

Parameter	PWR
Assembly power (MW/MTU)	32.3
Avg. fuel temperature (K)	898
Fuel density (kg/dm^3)	10.45
Coolant temperature (K)	577
Boron concentration (ppm)	600
Coolant density (kg/dm^3)	0.68
Cycle length (days)	345
Shutdown length (days)	20
Decay time (days)	365
Reactor lattice spacing (cm)	21.5036

The measured axial and horizontal burnup distributions referred to in SKBdoc 1193244 were not available for this review. However, the limiting curves shown in figures are useful.

The power is 15 MW/assembly. The unit required by SCALE 6.1 is specific power in MW/MTU (or kW/kgU). The mass per assembly is 464 kg according to Table 4. These values result in a specific power of 32.3 kW/kgU (specified as MW/MTU in SCALE).

10.1.5. PWR depletion models

The geometry model used for T5-depl (KENO V.a) and T6-depl (KENO-VI) was either a fuel assembly with mirror boundary or the STARBUCS automatic use of pre-calculated Westinghouse 17x17 OFA fuel cases. The T-depl model (NEWT) was based on a quarter symmetry model of a PWR-fuel assembly.

The reactor lattice model is the same as specified for the depletion in the EPRI benchmarks. A NEWT quarter-symmetry model is shown in Figure 27 (the colours are different than for the figures showing KENO V.a models). The KENO V.a and KENO-VI models were full 17x17 lattices, similar to Figure 25 but with a thin water region outside the lattice.

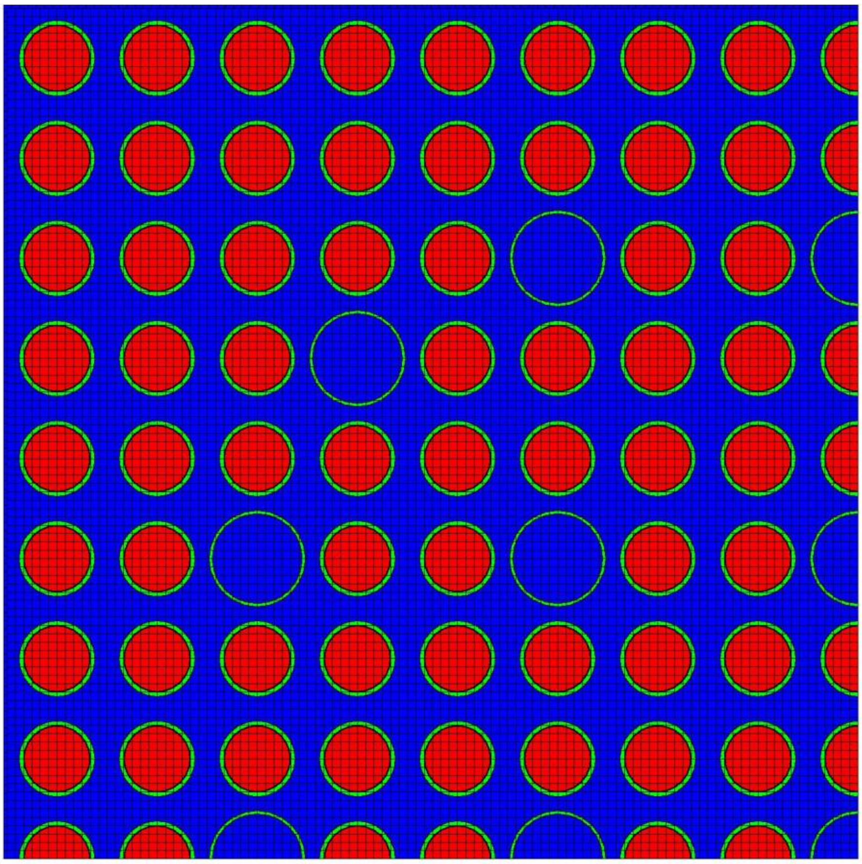

Figure 27: NEWT PWR fuel assembly lattice (1/4 symmetry)

10.1.6. PWR canister models

The geometry of the PWR canister is very simple to model with Monte Carlo codes like KENO V.a. and KENO-VI. Only the nominal case, with centred PWR assemblies in each insert position was calculated using a model shown in Figure 28.

The mass of fresh fuel uranium corresponding to the input model is 469 kg. This can be compared with the 464 kg specified by SKB in Table 4 for a slightly different fuel assembly.

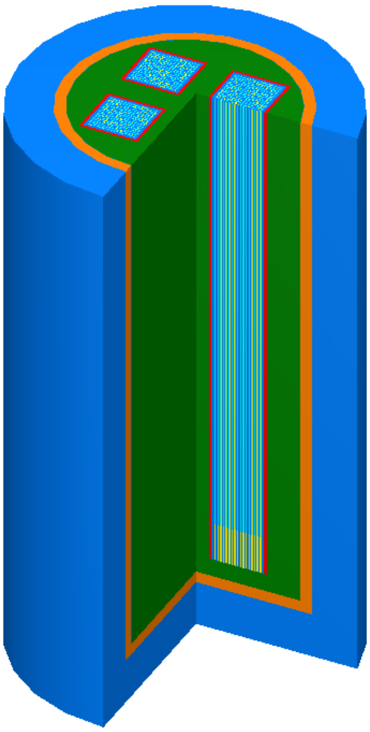

Figure 28: KENO V.a PWR fuel canister (top 1/2 and front right 1/4 plus a bit more cut away) with fuel

10.1.7. Calculation cases

For the PWR canister cases, calculations were first made for fresh fuel. Two KENO V.a calculations were made. The first had ENDF/B-VII.0 continuous energy cross sections while the second had ENDF/B-VII.0 238-group cross sections. In addition, one KENO VI calculation was made with ENDF/B-VII.0 continuous energy cross sections.

To determine the depletion influence on the fuel at low burnup values, many time steps were made during this period (can be seen in Figure 29). The burnup range was from 0 to 45 MWd/kgU. For each time step k_{eff} was calculated at hot reactor power conditions for a PWR fuel lattice with mirror reflection conditions. A constant power was assumed during the whole period, including during start-up.

Depletion calculations were made with three TRITON sequences (T-depl, T5-depl and T6-depl) referred to in chapter 10.1.5. Three burnup values were selected: 15 MWd/kgU, 30 MWd/kgU and 45 MWd/kgU. The full actinide and fission product set (more than 300 nuclides) were selected for the depletion.

For the PWR canister, only the nuclides selected by SKB for the actinide and fission product cases were accounted for (conservative). KENO V.a and KENO-VI calculations were made for fuel having an average burnup over the whole assembly. STARBUCS was used with such a flat burnup distribution but also with the built-in 18-node axial burnup distribution.

ENDF/B-VII.0 cross-sections in the 238-group format were used for the depletion calculations (collapsed to a built-in 49-group library in NEWT) and for copper canister calculations. In addition, ENDF/B-VII.0 continuous energy format cross-sections were used to calculate the PWR-canister using KENO V.a, KENO VI and STARBUCS.

10.1.8. Calculation results

The results for the fresh fuel PWR canister cases are shown in Table 30 together with results for depleted fuel.

The use of different codes and different cross-sections here doesn’t make the calculations independent. The codes are very different; KENO VI is not a later version of KENO V.a but a separate code. However, they have been developed by the same organisation. The cross-section libraries and treatment are identical for the two codes. The two different cross-section libraries are both based on ENDF/B-VII.0.

The fresh fuel results show small differences. This is expected since the system is well moderated. The 238-group library should perform well for such systems. The known problems with some of the SCALE 6.1 continuous energy cross-sections have to some extent been resolved in SCALE 6.1.1 and the remaining problems only appear to involve faster systems.

MCNP5 has not yet been used but allows independent calculations. The PWR canister is easy to model in MCNP5.

The infinite fuel assembly lattice depletion results from the three methods are very close, as expected from the preliminary validation work. Figure 29: shows the calculated relationship between burnup and k_{inf} using T5-depl (KENO V.a) for hot full power conditions (high temperatures). The results using T6-depl (KENO VI) and T-depl (NEWT) are very similar and are not shown. Note the extra sharp drop in k_{inf} at start-up due to the instant build-up of fission product equilibrium concentrations at full power.

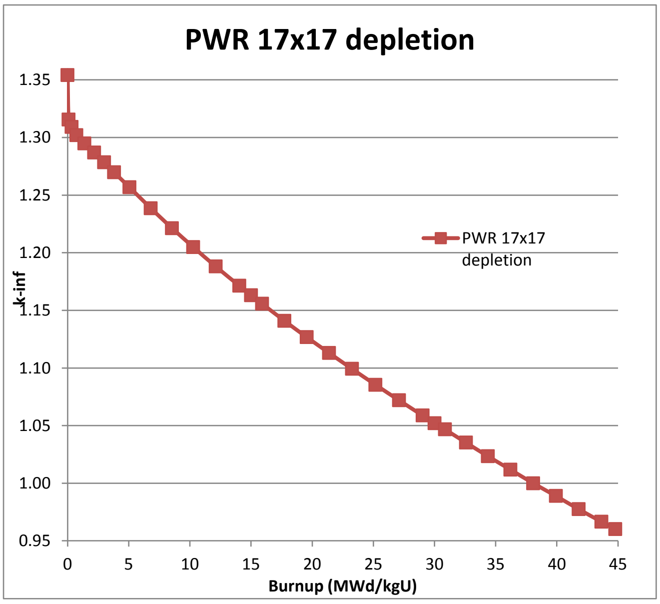

Figure 29: PWR 17x17 assembly depletion k_{inf} under hot full power reactor conditions

For the PWR canister cases there were no major differences between different libraries and codes, with one exception: the STARBUCS k_{eff} result for an average burnup of 45 MWd/kgU and one axial region is significantly lower (0.798) than the result from KENO V.a. (0.825). The result for STARBUCS with 18 axial regions is also much higher (0.830) than for the flat distribution.

Results for the PWR canister cases with fresh and depleted fuel are shown in Table 30: and in Figure 30. The results for the STARBUCS calculation of 45 MWd/kgU case using a single axial region appears to be low. The reason is not known.

Table 30: PWR canister with depleted fuel

Case	Axial burnup distribution	SCALE 6.1.1 sequence	Code for k_{eff}	ENDF/B-VII.0 cross-sections	k_{eff}	σ
Fresh fuel	Uniform	CSAS5	KENO Va	Cont. energy	1.0818	0.00025
Fresh fuel	Uniform	CSAS5	KENO Va	238-groups	1.0828	0.00030
Fresh fuel	Uniform	CSAS6	KENO VI	Cont. energy	1.0817	0.00025
15 MWd/kgU	Uniform	CSAS5	KENO Va	Cont. energy	0.9763	0.00025
15 MWd/kgU	Uniform	STARBUCS	KENO Va	238-groups	0.9741	0.00022
15 MWd/kgU	18 nodes	STARBUCS	KENO Va	238-groups	0.9901	0.00025
30 MWd/kgU	Uniform	CSAS5	KENO Va	Cont. energy	0.8945	0.00025
30 MWd/kgU	Uniform	STARBUCS	KENO Va	238-groups	0.8847	0.00022
30 MWd/kgU	18 nodes	STARBUCS	KENO Va	238-groups	0.8925	0.00023
45 MWd/kgU	Uniform	CSAS5	KENO Va	Cont. energy	0.8253	0.00025
45 MWd/kgU	Uniform	STARBUCS	KENO Va	238-groups	0.7980	0.00019
45 MWd/kgU	Uniform	STARBUCS	KENO Va	Cont. energy	0.7941	0.00022
45 MWd/kgU	18 nodes	STARBUCS	KENO Va	238-groups	0.8299	0.00024

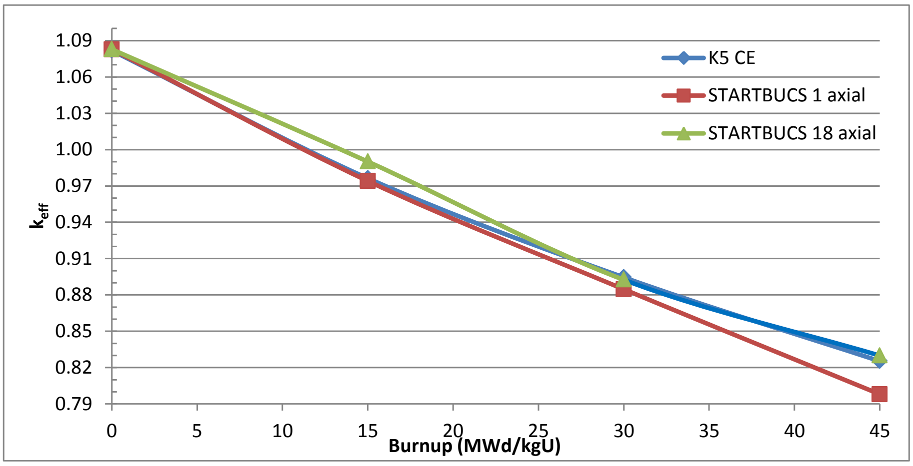

Figure 30: Calculation results for PWR fuel canister using different methods and models

11. Comparisons of EMS and SKB results

The agreement between similar cases is very good considering the different methods and calculation models.

The SKB result for a water-reflected PWR fuel canister with fresh fuel is 1.0868 while the corresponding EMS result is 1.0818. There are fuel assembly and other known parameter differences that explain a higher result by SKB.

Since the SKB values for PWR fuel (Table 41 in SKBdoc 1193244) originally intended to be used in the comparison are incorrect, Figure 20 of this Technical Note (Figure 16 of SKBdoc 1193244) is used here for comparison with EMS results.

First, it must be observed that the SKB results in Figure 20 are based on bounding parameter values while the EMS results are based on nominal parameter values. The SKB result for the water-reflected fresh fuel case is 1.1041. The difference in k_{eff} to the EMS result is about 0.022 (SKB results being higher).

For depleted fuel in PWR canisters with fuel having uniform axial burnup distributions, the agreement between SKB and EMS CSAS5 results is quite good, after accounting for the model difference 0.022 mentioned above. For the STARBUCS result, the agreement is good except for 45 MWd/kgU where the EMS result is unexpectedly low:

- For 15 MWd/kgU the EMS k_{eff} results are about 0.975. The corresponding SKB result is slightly higher than 1.000.
- For 30 MWd/kgU the EMS k_{eff} results are about 0.890. The corresponding SKB result is about 0.915.
- For 45 MWd/kgU the EMS k_{eff} result using CSAS5 is 0.825 while the STARBUCS value is 0.798. The corresponding SKB result is about 0.84.

The STARBUCS results by SKB and EMS for the 18 axial (node) burnup distribution models agree well:

- For 15 MWd/kgU the EMS k_{eff} result is 0.990. The corresponding SKB result is slightly higher than 1.000.
- For 30 MWd/kgU the EMS k_{eff} result is 0.893. The corresponding SKB result is about 0.915.
- For 45 MWd/kgU the EMS keff result is 0.830. The corresponding SKB result is about 0.84.

The EMS results indicate larger and positive differences (+0.015, +0.003 and +0.032 respectively for the three burnups) between the 18 axial node model and the uniform model cases than what the SKB results for the “end effect” of maximum F in Figure 21 indicate (-0.005, -0.005 and ±0.000 respectively for the three burnups).

The EMS k_{eff} calculations based on new depletion calculations (as opposed to STARBUCS) were only made for the uniform axial burnup distribution. The additional work required to generate cross-sections for these 18 nodes for three different burnup values would be justified later if more information on the measured burnup distribution becomes available.

12. The Consultant’s assessment

The SKB criticality safety evaluation of an intact copper canister for PWR fuel has been reviewed by EMS for SSM. The result is that the selection and evaluation of parameters that may be significant for k_{eff} appear to be complete and correct. The calculated results, accounting for known parameter differences, have been confirmed. This conclusion applies both to fresh fuel and to depleted fuel.

Some information that could reduce the differences and uncertainties is missing to the reviewer and possibly to SKB. There also appears to be some errors (editorial) in the information provided in the safety documentation. Missing or questionable data include:

- Table 41 of SKBdoc 1193244 is supposed to contain both BWR and PWR results. The PWR data specified are identical to the BWR data and that could not be correct. The PWR data was needed during the final step of the review to compare SKB results with EMS review results. Similar information can be approximately obtained from other data in the report but an updated Table 41 is needed;
- The quality control of the canisters is important. The reviewed information is several years old and there may be more accurate information available. In particular the PWR insert channel geometry variations after casting and the centre-centre spacing variations, as well as potential voids in the nodular iron, may be important for reducing the uncertainties;
- Table 2 and Table 22 in this Technical Note show differences between some fuel parameter values in different SKB documents. They are probably not significant but some effort should be made by SKB to obtain consistent information;
- The actual reactor geometry needs to be specified to some extent to model the fuel depletion by calculations. The reactor geometry model used in SKBdoc 1193244 depletion calculations should also be specified;
- Actual burnup determination data and how they are obtained are missing;

- Additional reactor operating conditions needed to simulate depletion are missing. An example is presence of control rods (pointed out by SKB);
- Measured axial burnup profiles are missing;
- Measured horizontal burnup profiles are missing;
- More detailed fuel assembly end region specifications are needed;
- More data on PWR-MOX assemblies and their reactor depletion parameters are needed. See e.g. chapter 3.1 (geometry) and 4 (last paragraph). SKB is well aware of this according to the application documentation.

The degree of detailed review calculations required in future reviews depends on the safety margins of the final specifications. It appears as if the SKB criticality safety document has covered all essential parameters. The k_{eff} sensitivities to those parameter variations have not been independently confirmed by the reviewer but they appear to be credible.

The EMS selection of methods for criticality safety assessment of the SKB application for final disposal appears to be well suited to the task. There are many options and a high degree of accuracy may be obtained. High accuracy may come at the expense of considerable extra work and long computer calculations. The simplified STARBUCS sequence in SCALE appears to give adequate results.

General validation of EMS burnup credit methods has been started, with preliminary results sufficient for this review. Both OECD/NEA calculation benchmarks (supported by several established methods and users) as well as the EPRI depletion reactivity benchmarks have been calculated with good results. This is an on-going project that involves all burnup credit and burnable absorber credit.

On-going EPRI and OECD/NEA work on converting the EPRI benchmarks to IRPhEP Handbook benchmark experiments is followed directly by the author as an independent reviewer of the proposal. A draft evaluation will be published in the spring 2013 edition of the Handbook. EMS results from SCALE 6.1.1 compare well with Studsvik CASMO 5/SIMULATE 3 results for the proposed benchmarks. This provides confidence to the author even though the results can't be published yet.

The Finnish TVO is working on development of combined benchmarks for depletion and burnup credit for BWR fuel. The benchmarks are based on cold critical reactor measurements of control rod movements. It is currently unknown whether they will become available outside TVO.

The recent (September 2012) NRC SFST ISG-8 Rev. 3 is a significant update to the revision 2 that is referred to by SKB in the application documents (e.g. SKBdoc1193244). The validation guide is of interest as a complement to the EPRI benchmarks. This Guide has not been studied in connection with the current review but may be of particular interest in assessment of misloading events.

Concerning future review of BWR burnup credit and burnable absorber credit, a recent (September 2012) initiative by Japan to add a new Phase IIIC to the OECD/NEA burnup studies has been approved. Results are to be provided before the end of February 2013. The Phase IIIC is briefly described in Appendix A.

The EMS validation experience suggests that the early depletion (low burnup) may be more complicated than later depletion (high burnup). This is particularly true for fuel with burnable absorbers (IFBA fuel rods in PWR and gadolinium rods both in

BWR and PWR fuel). The reactivity peak caused by gadolinium for BWR fuel has a correspondence in a reactivity peak for some IFBA fuel for PWR. This is not a concern for the current review since IFBA credit is not applied.

13. References

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21. “The FR0 Critical Experiments - U(20) Metal in Fast and Intermediate Spectra”, Dennis Mennerdahl, ICNC 2011, Edinburgh, September 2011.

Appendix A. OECD/NEA/NSC/WPNCS

The main report reference 9 contains a link to OECD/NEA reports that provide published specifications and results for the phases I-B, III-B, IV-B and II-D referred to below. The specifications for those phases as well as for the on-going phase III-C are available at the following web site;

http://www.oecd-nea.org/science/wpncs/buc/specifications/

A.1. Phase I-B

Reference:"OECD/NEA Burnup Credit Calculational Criticality Benchmark Phase I-B Results", M. D. DeHart, M. C. Brady, C. V. Parks, NEA/NSC/DOC(96)-06 (ORNL ORNL-6901), June 1996.

The main purpose of this comparison benchmark was to find differences in nuclide inventories for three cases A, B and C. The influence on the neutron multiplication factor was added as an appendix prepared by CSN (Spain).

Figure A.1 shows a NEWT geometry model with materials and grid lines. Later results demonstrate that more grid lines should be used to get accurate results.

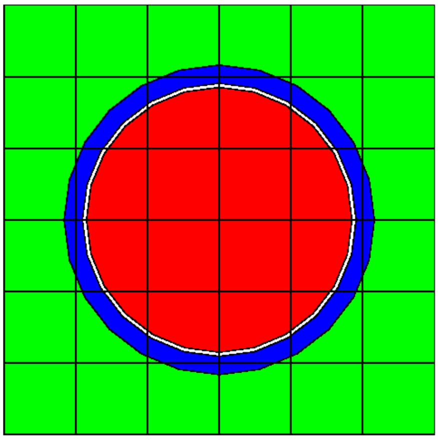

Figure A.1: NEWT geometry

The k_{inf} and depletion reactivity values calculated with SCALE 6.1.1 t-depl sequence are presented in Table A.1 and are compared with the average OECD results. The measured results are based on calculations with nuclide inventories measured from samples.

Table A.1: SCALE 6.1.1 t-depl results

Id	Method	EMS k_{inf}		OECD depleted kinf	
IB	SCALE 6.1.1, 238-g E-VII.0	Fresh	Depleted	Measured	Average
IB-A	t-depl (NEWT)	1.30273	0.99905	1.01842	1.01721
IB-B	t-depl (NEWT)	1.30273	0.91990	0.92193	0.91776
IB-C	t-depl (NEWT)	1.30273	0.87567	0.85320	0.85474

Calculation results for the sequences t-depl-1D (XSDRNPM/S), t5-depl (KENO V.a) and t6-depl (KENO-VI) were similar. The NEWT results depend on the calculation parameters such as cylinder approximation, grid mesh and S_n-order,

The calculated nuclide inventories appear to be reasonably in agreement with the OECD results, considering the large spread in those results.

A.2. Phase III-B

Reference:"OECD/NEA Burnup Credit Criticality Benchmarks Phase IIIB: Burnup Calculations of BWR Fuel Assemblies for Storage and Transport", H. Okuno, Y. Naito, K. Suyama, NEA/NSC/DOC(2002)2 (JAERI-Research 2002-001), 2002

There are three cases: A, B and C.

Calculation methods: SCALE 6.1.1, sequences T-DEPL, T-DEPL-1D, T5-DEPL and T6-DEPL. The cross-sections come from the 238-group ENDF/B-VII.0 library.

Figure A.2 shows a NEWT geometry model with materials and grid lines. The many colours were generated by NEWT without influence of the author. The number of materials could have been reduced significantly (e.g. the same water in all fuel cells). Later results demonstrate that more grid lines should be used to get accurate results.

Results for cases A, B and C are shown in Figures A.3, A.4 and A.5. It seems as if the peak reactivity obtained by SCALE 6.1.1 is higher. The few points calculated in the OECD study makes the fitted curve very unreliable in this area.

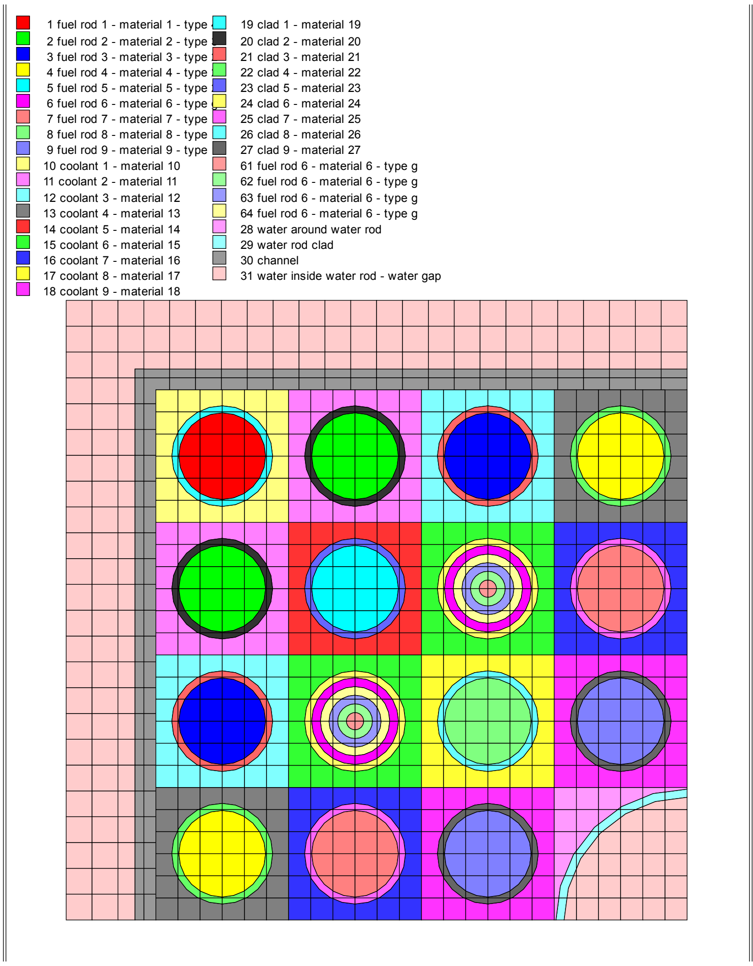

Figure A.2: NEWT geometry for OECD Phase IIIB benchmark

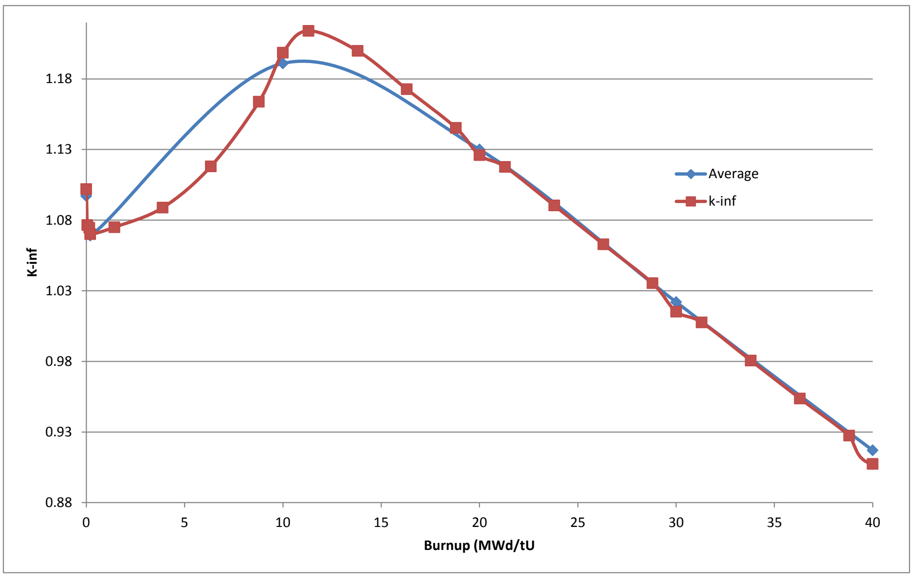

Figure A.3: Case A - NEWT results compared with Phase IIIB participants' average.

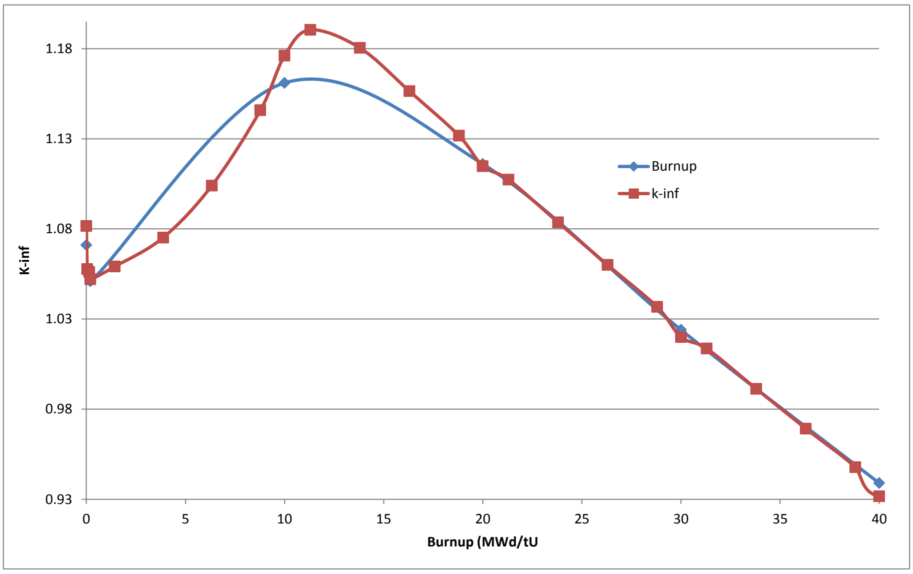

Figure A.4: Case B - NEWT results compared with Phase IIIB participants' average.

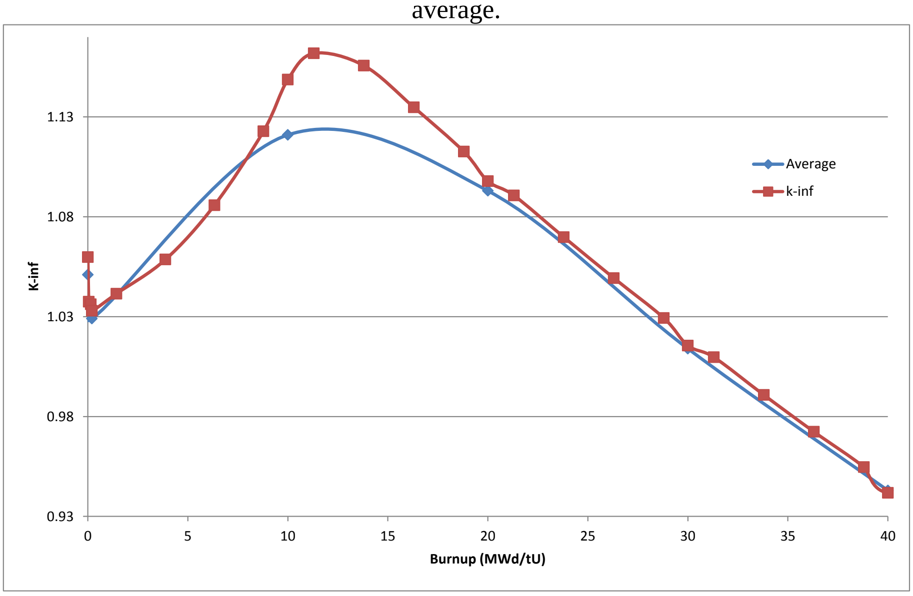

Figure A.5: Case C - NEWT results compared with Phase IIIB participants' average.

A.3. Phase IV-B

Reference: "Burn-up Credit Criticality Benchmark PHASE IV-B: Results and Analysis of MOX Fuel Depletion Calculations", G. J. O'Connor, Peng Hong Liem, NEA/NSC/DOC(2003)4, April 2003

There are three calculation models in the Phase IV-B exercise:

1. A supercell with a PWR MOX assembly together with three PWR UO2 fuel assemblies with periodic boundaries.

2. A PWR MOX-only core representation with reflective boundary conditions
3. A simple MOX pin cell, using the average MOX fuel composition and moderation.

For each model there are two plutonium isotope distributions; one reactor-grade and one weapons-grade. The number of cases is thus six.

Calculation methods: SCALE 6.1.1, sequences T-DEPL has been used to start validation using these calculation benchmarks. The cross-sections come from the 238-group ENDF/B-VII.0 library.

Figure A.6 shows a NEWT geometry model with materials and grid lines for the most complicated model (1).

The large supercell makes calculations with NEWT very slow. Preliminary results appear to be good but the calculations have been postponed.

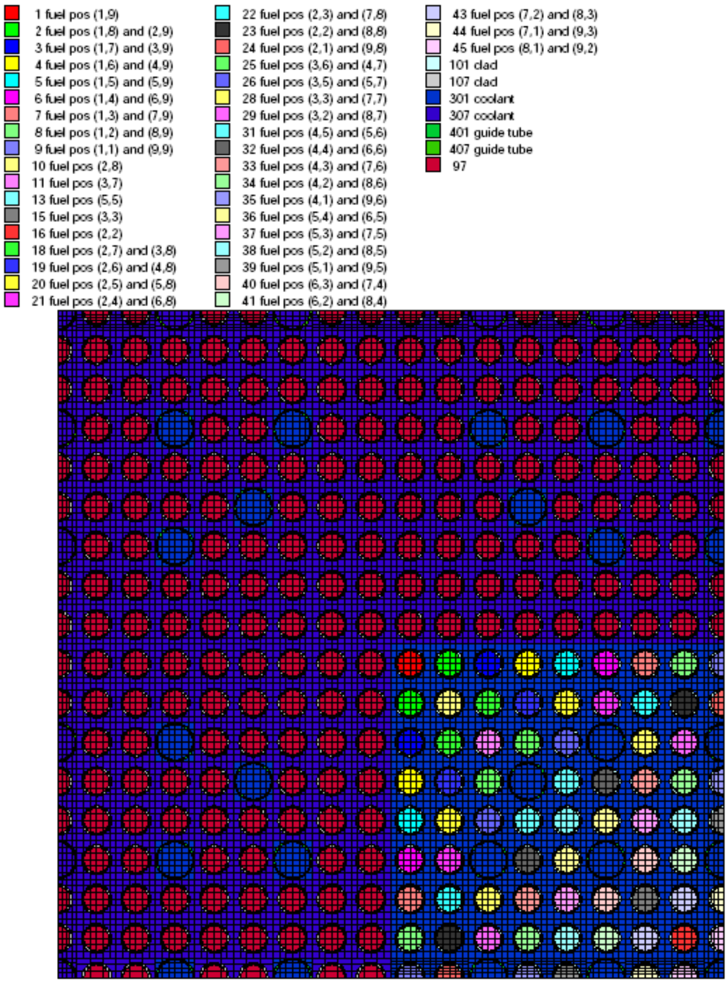

Figure A.6: NEWT geometry for OECD Phase IVB benchmark

A.4. Phase II-D

Reference: ”Burn-up Credit Criticality Benchmark - Phase II-D PWR-UO2 Assembly Study of Control Rod Effects on Spent Fuel Composition”, A. Barreau, NEA No. 6227, 2006.

The Phase IID benchmark exercise involved a study of the influence of control rod (CR) insertion on the spent fuel composition and on k_{eff} for a PWR UO_2 assembly. The nuclide inventory calculations are not discussed here.

There are 12 depletion cases and k_{inf} is requested for each both with and without fission products (a and b added to the case numbers). In addition there are two cases 13 and 14 with “imposed fuel inventory” which don’t include depletion calculations. This also applies to case 15 which is fresh fuel k_{inf}.

A short description of the cases is presented in Table A.2. “SD” stands for standard deviation of the average participant results.

An EMS input model for SCALE 6.1.1 and NEWT is shown in Figure A.7. These calculations have not been completed.

Table A.2: Phase II-D case descriptions

Case	Burnup (MWd/kgU)	Fission products?	Control rod insertion (MWd/kgU)		Cooling	Average OECD/NEA results	
		Yes/No	Start	End	Hours/Years	k_{inf}	SD
1a	30	N	No CRs	No CRs	0	1.17036	0.00611
1b	30	Y	No CRs	No CRs	0	1.09694	0.00681
2a	30	N	0	30	0	1.22263	0.00472
2b	30	Y	0	30	0	1.14648	0.00547
3a	45	N	No CRs	No CRs	0	1.08166	0.00962
3b	45	Y	No CRs	No CRs	0	0.99124	0.01025
4a	45	N	0	45	0	1.17550	0.00358
4b	45	Y	0	45	0	1.08005	0.00839
5b	45	Y	0	15	0	1.01167	0.00940
6b	45	Y	15	30	0	1.01905	0.00972
7b	45	Y	30	45	0	1.04355	0.00908
8b	45	Y	0	30	0	1.03868	0.00908
9a	30	N	No CRs	No CRs	5	1.15595	0.00597
9b	30	Y	No CRs	No CRs	5	1.06710	0.00616
10a	30	N	0	30	5	1.20647	0.00463
10b	30	Y	0	30	5	1.11603	0.00601
11a	45	N	No CRs	No CRs	5	1.05746	0.00925
11b	45	Y	No CRs	No CRs	5	0.94356	0.01023
12a	45	N	0	45	5	1.14914	0.00717
12b	45	Y	0	45	5	1.03460	0.01035
13b	30	Y	No CRs	No CRs	0	0.93861	0.00223
14b	30	Y	No CRs	No CRs	0	1.02619	0.00235
15	Fresh	-	-	-	-	1.33986	0.00169

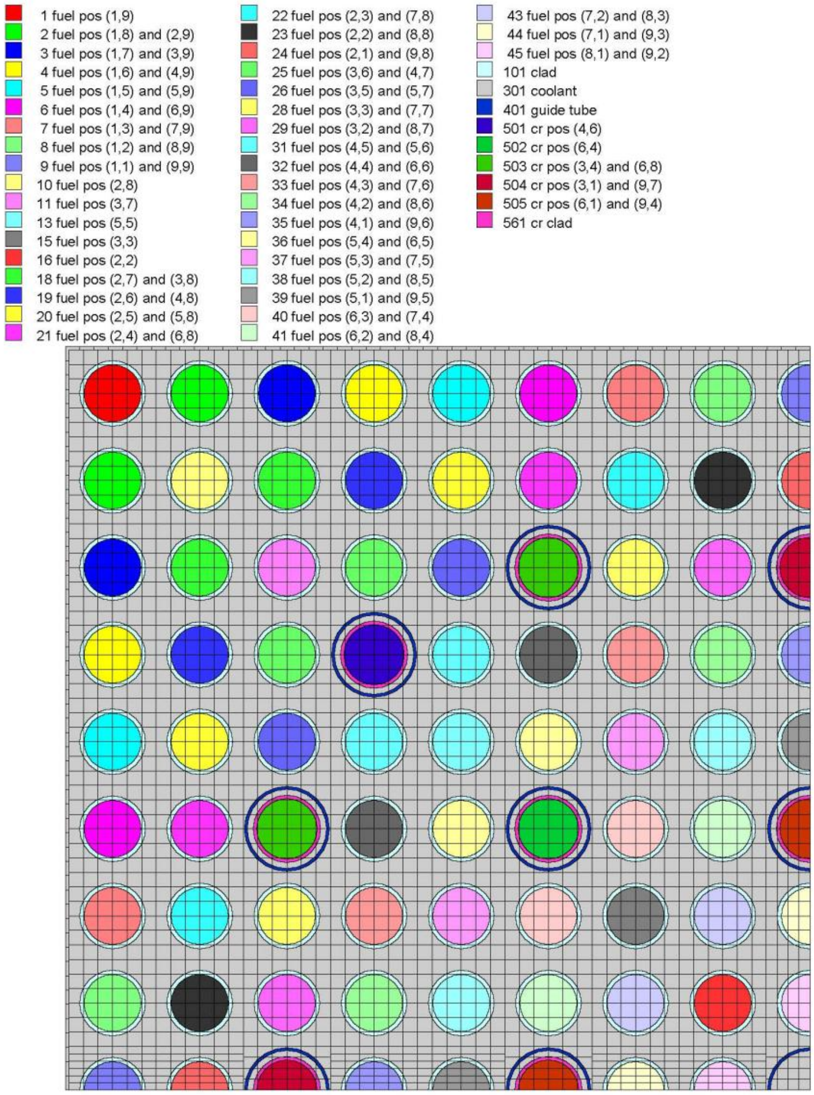

Figure A.7: Phase IID case 4 geometry model for NEWT

A.5. Phase III-C

Reference: “OECD/NEA Burnup Credit Criticality Benchmark Phase IIIC - Nuclide Composition and Neutron Multiplication Factor of BWR Spent Fuel Assembly for Burnup Credit and Criticality Control of Damaged Nuclear Fuel”, K. Suyama, Y. Uchida, T. Ito, T. Miyaji, September 2012.

This Phase is similar to Phase IIIB but has a more modern design. The Japanese proposal is related to the Fukushima accident. A NEWT input model is shown in Figure A.8.

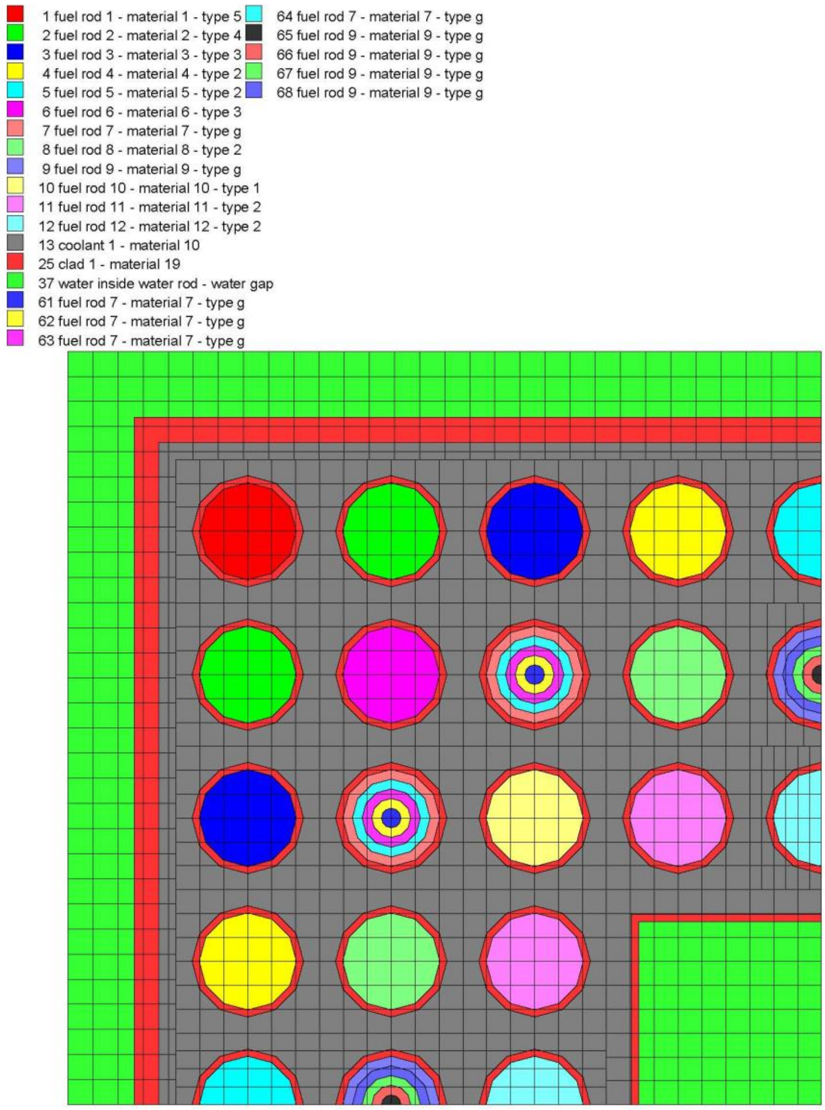

Figure A.8: Phase IIIC geometry model for NEWT

Some calculations have been made to check the input. Participation in the OECD/NEA study is planned and the final deadline for supplying the data at the end of February should be met.

It is expected that this Phase will inform the participants about current calculation methods and techniques to obtain good results with reasonable calculation times.

Appendix B. EPRI benchmarks

The main report reference 18 contains a link to the EPRI report that provides published specifications for these benchmarks.

B.1. Introduction

The EPRI benchmarks published in 2011 is based on a combination of measurements and well validated calculation results. Since the validation for the method (CASMO-5) is proprietary (Studsvik), a validation based on publicly available information was made.

The measurements involve hot full power conditions of 44 cycles at four PWRs in the U.S.A. The EPRI benchmarks are for cold conditions after various decay times. Benchmarks for eleven different fuel lattice types were published. The hot full power measurements are currently being evaluated as a benchmark experiment for the OECD IRPhE Handbook. A draft will be published in the 2013 version of the handbook. Appendix C contains more about this evaluation.

During work with the IRPhE evaluation, some concerns with the EPRI benchmarks have been identified by the author. A major problem is the definition of depletion reactivity for fuel with burnable absorbers. It is defined as the difference between the k_{inf} value for the depleted fuel with burnable absorber and the k_{inf} value for fresh fuel without burnable absorbers. This can lead to application problems, in particular when one method is used for depletion calculations and another method is used for calculation of the k_{inf} value.

In spite of some concerns, the EPRI benchmarks appear to be the most appropriate publicly available sources for validation of burnup credit for PWR fuel.

There are 11 benchmark lattice types. Each of the 11 lattice types has benchmark data for 0, 100 hour, 5 year, and 15 year cooling time. Each case was done for 6 burnup values; 10, 20, 30, 40, 50, and 60 MWd/kgU burnup. The benchmark reference lattice does not include burnable absorbers even if they are present during depletion.

Cases 1-10 are depleted with a power density of 104.5 W/cm^3 (38.1 kW/kgU) while case 11 is depleted at 156.75 W/cm^3 (150 % of nominal power density).

The number of cases is 264 (11 x 4 x 6).

Table B.1 contains a summary of the cases.

Table B.1: EPRI benchmark lattice types

Type	Specifications
1	3.25 wt.% enrichment
2	5.00 wt.% enrichment
3	4.25 wt.% enrichment
4	Off-nominal pin diameter
5	20 WABA rods
6	104 IFBA rods
7	104 IFBA plus 20 WABA rods
8	High boron = 1500 ppm
9	Hot rack coolant/fuel = 338.7K
10	High rack boron = 1500 ppm
11	High power, coolant/fuel temp

The results are depletion reactivities for lattice types, i.e. differences between k_{inf} values for fresh fuel for fuel at a specific burnup and cooling time. At this time, only lattice type 1 has been calculated.

B.2. Case 1

This case is standard PWR fuel without burnable absorbers. The ^{235}U enrichment is only 3.25 wt.%. Figure B.1 shows the quarter PWR assembly as modelled for NEWT.

EMS results using SCALE 6.1.1 with the T-depl sequence based on NEWT are presented in Table B.2. The results are considered to be very good.

Table B.2: EMS results for lattice type 1 using NEWT

Case	Burnup (MWd/kg U)	Cooling time (hours/years)	Depletion reactivity		Bias
			Benchmark	EMS- NEWT	
1	10	0	-0.1779	-0.17712	0.00005
2	10	100 hours	-0.1329	-0.13326	-0.00036
3	10	5 years	-0.1370	-0.13776	-0.00076
4	10	15 years	-0.1422	-0.14331	-0.00111
5	20	0	-0.2754	-0.27427	0.00009
6	20	100 hours	-0.2339	-0.23434	-0.00044
7	20	5 years	-0.2471	-0.24793	-0.00083
8	20	15 years	-0.2655	-0.26657	-0.00107
9	30	0	-0.3589	-0.35758	0.00009
10	30	100 hours	-0.3211	-0.32144	-0.00034
11	30	5 years	-0.3447	-0.34544	-0.00074
12	30	15 years	-0.3768	-0.37773	-0.00093
13	40	0	-0.4302	-0.42992	0.00028
14	40	100 hours	-0.3956	-0.39576	-0.00016
15	40	5 years	-0.4284	-0.42901	-0.00061
16	40	15 years	-0.4720	-0.47279	-0.00079
17	50	0	-0.4873	-0.48695	0.00035
18	50	100 hours	-0.4554	-0.45543	-0.00003
19	50	5 years	-0.4951	-0.49569	-0.00059
20	50	15 years	-0.5471	-0.54794	-0.00084
21	60	0	-0.5300	-0.52951	0.00049
22	60	100 hours	-0.5002	-0.49999	0.00021
23	60	5 years	-0.5445	-0.54501	-0.00051
24	60	15 years	-0.6021	-0.60285	-0.00075

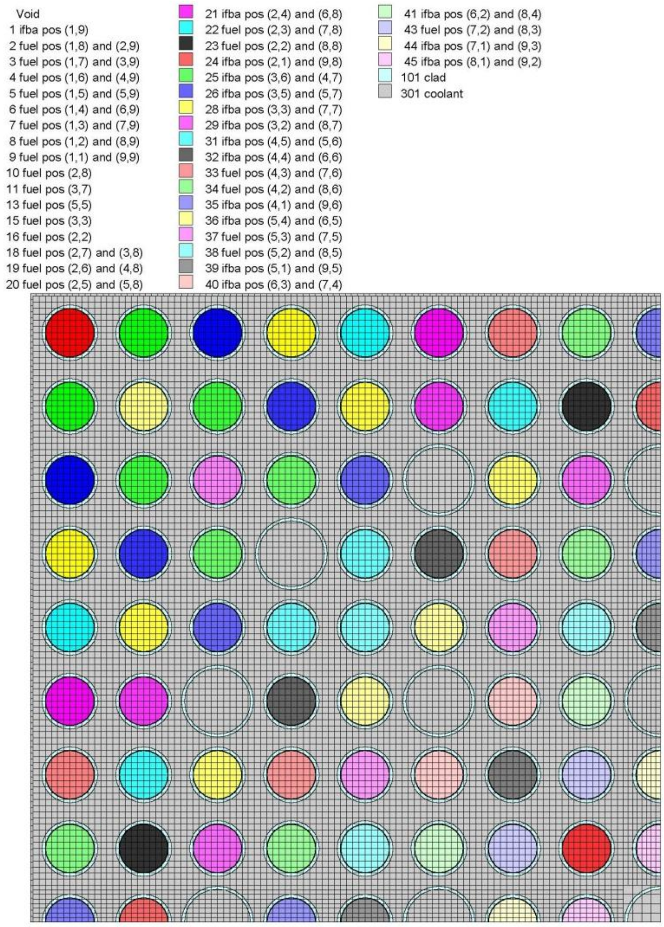

Figure B.1: EPRI benchmark case 1 geometry model for NEWT

Appendix C. IRPhE Handbook benchmark

Reference: There is not yet any published reference on this proposed benchmark experiment. The geometry is essentially identical to one of the EPRI benchmarks in Appendix B.

EPRI has proposed a depletion reactivity benchmark for the OECD/NEA IRPhE Handbook (reactor physics) based on the EPRI depletion reactivity benchmarks published in 2011 (see appendix B).

This work is ongoing, but a draft will be included in the spring 2013 edition of the Handbook. Since the EPRI benchmarks may rely too much on validated calculations to be directly appropriate for the Handbook (measurements), significant changes are being made.

As one of the external reviewers for this evaluation and as a participant in the IRPhE, I have access to the current and previous drafts of the evaluation. Since it is a working document, the exact specifications and results will not be included here. There may still be changes in the IRPhE draft evaluation expected to be published in 2013.

The review of this evaluation has involved use of SCALE 6.1.1 for hot full power depletion calculations. The experience is valuable and has not yet shown signs of any significant error in the method.

Appendix D. Coverage of SKB reports

Table D.1:

Reviewed report	Reviewed sections	Comments
SKBdoc 1091554 Version 3.0, ”Säkerhetsredovisning för drift av slutförvarsanläggning för använt kärnbränsle (SR-Drift) kapitel 3 - Krav och konstruktionsförutsättningar”,	All that was considered relevant for intact copper canisters with PWR fuel	
SKBdoc 1091152 Version 3.0, ”Säkerhetsredovisning för drift av slutförvarsanläggning för använt kärnbränsle (SR-Drift) - Inventering av yttre och inre händelser för slutförvarsanläggningen”	All that was considered relevant for intact copper canisters with PWR fuel	
SKBdoc 1091141 Version 3.0, ”Säkerhetsredovisning för drift av slutförvarsanläggning för använt kärnbränsle (SR-Drift) kapitel 8 – Säkerhetsanalys”	All that was considered relevant for intact copper canisters with PWR fuel	
SKB TR-10-13, “Spent nuclear fuel for disposal in the KBS-3 repository”	All that was considered relevant for intact copper canisters with PWR fuel	
SKBdoc 1193244 version 4.0, “Criticality safety calculation of disposal canisters”	All that was considered relevant for intact copper canisters with PWR fuel	
SKB TR-10-14, “Design, production and initial state of the canister”	All that was considered relevant for intact copper canisters with PWR fuel	
SKB TR-10-52, “Data report for the safety assessment SR-Site”	All that was considered relevant for intact copper canisters with PWR fuel	

2013:16

The Swedish Radiation Safety Authority has a comprehensive responsibility to ensure that society is safe from the effects of radiation. The Authority works to achieve radiation safety in a number of areas: nuclear power, medical care as well as commercial products and services. The Authority also works to achieve protection from natural radiation and to increase the level of radiation safety internationally.

The Swedish Radiation Safety Authority works proactively and preventively to protect people and the environment from the harmful effects of radiation, now and in the future. The Authority issues regulations and supervises compliance, while also supporting research, providing training and information, and issuing advice. Often, activities involving radiation require licences issued by the Authority. The Swedish Radiation Safety Authority maintains emergency preparedness around the clock with the aim of limiting the aftermath of radiation accidents and the unintentional spreading of radioactive substances. The Authority participates in international co-operation in order to promote radiation safety and finances projects aiming to raise the level of radiation safety in certain Eastern European countries.

The Authority reports to the Ministry of the Environment and has around 270 employees with competencies in the fields of engineering, natural and behavioural sciences, law, economics and communications. We have received quality, environmental and working environment certification.

Strålsäkerhetsmyndigheten
Swedish Radiation Safety Authority

SE-171 16 Stockholm
Solna strandväg 96

Tel: +46 8 799 40 00
Fax: +46 8 799 40 10

E-mail: registrator@ssm.se
Web: stralsakerhetsmyndigheten.se