71-3034

U.S. Department of Transportation

Pipeline and Hazardous Materials Safety Administration

1200 New Jersey Ave, S.E.
Washington, D.C. 20590

JUL 09 2015

Michele Sampson, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety and Safeguards (NMSS)
U.S. Nuclear Regulatory Commission
11545 Rockville Pike
Mail Stop T4B34
Rockville, MD 20852-2738

Dear Ms. Sampson:

In accordance with the Memorandum of Understanding between our agencies, I request that you review the enclosed French Certificate of Approval No. F/313/B(U)F-96, Revision Jbb, for the TN-BGC1 package and make a recommendation concerning our revalidation of the package for import and export use.

On November 19, 2013 I requested that you review this package, limited to content number 11 (non- irradiated uranium-bearing materials in any solid form) and content number 26 (TRIGA fuel) from the French certificate. In your February 11, 2015 letter you advised me that you had terminated your review of this package design because you had not received the information you requested in your January 30, 2014 request for additional information (Docket No. 71-3034, TAC No. L24858). I have received the enclosed updated request from Areva TN which includes, as Appendix 2, their responses to your January 30, 2014 request for additional information. They are requesting approval of content 11 (excepting 11(h), thereby adding 11 (a), (b), (c), (g) to contents we have previously authorized), with the mass limits shown in their attached Appendix 1, and content 26.

To assist you in your review, technical data to support this review is contained on the enclosed disc.

57 NPPR files NMSS01
1 prop file

Since our applicant has indicated a need to use this package for a shipment tentatively scheduled for the beginning of 2016, I request you provide an estimate of the time needed to complete your review. If you have any questions or need any additional safety information, please feel free to contact Michael Conroy of my staff at (202) 366-3597 or via email at Michael.Conroy@dot.gov.

Sincerely,

Richard Boyle
Division of Engineering and Research
Office of Hazardous Materials Safety

Enclosures

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For the attention of **Mr. BOYLE**

Montigny-le-Bretonneux, 25th June 2015

O/Ref.: CEX-15-00117104-001

Subject: **TN-BGC 1 package**

Application for American validation of the French certificate of approval F/313/B(U)F-96 (Jbb) for the package consisting of the TN-BGC1 packaging loaded with contents n°11 or n°26.

References:

<1> Safety Analysis Report CEA DSN/STMR/LEPE/TNBGC1 DSEM 0600 Rev. 02 of October 11th, 2012;

<2> TN-BGC1 French Certificate of Approval No. F/313/B(U)F-96 (Jbb);

<3> TN-BGC1 USA Certificate of Approval No. USA/0492/B(U)F-96 Rev. 16;

<4> Request for additional information for revalidation of model No. TN-BGC1 (French competent authority certificate of approval No. F/313/B(U)F-96 Rev. Jbb), Docket No. 71-3034, TAC No. L24858 of January 30th 2014;

<5> Application for the revision of certificate USA/0492/B(U)F-96 revision 16 from AREVA TN : ref. CEX-15-00117104-000 of February 23rd, 2015;

<6> Criticality code validation for TN-BGC1 - contents 11a, 11b, 11c and 11g, ref. TN-BGC1-0600 rev. B of June 12th 2015;

<7> Criticality analysis for TN-BGC1- contents 11a, 11b, 11c and 11g ref. TN-BGC1-0601 rev.1 of June 16th 2015;

Dear Mr. Boyle,

The TN-BGC 1 package is provided for the transport and storage of fissile material in very varied forms such as ingots of plutonium or metallic uranium, powders consisting of plutonium oxide or highly enriched uranium, and liquids such as uranyl nitrate.

By its decision of October 10th 2013 and on the basis of the safety analysis report <1>, the French Competent Authority issued the approval certificate in reference <2> expiring on June 30th 2018, for the package consisting in the TN-BGC 1 packaging loaded with the following contents of fissile material:

- Content n°2: Non-irradiated uranium oxide powder ;
- Content n°4: Ingots of non-irradiated metallic uranium;
- Content n°7: Non irradiated uranium oxide pellets, section of rods or rods;
- Content n°11: Non irradiated uranium-bearing materials in any solid form;
- Content n°26: TRIGA fuel;
- Content n°40: Uranyl nitrate;
- Content n°41: U-Zr fuel plates;
- Content n°42: Mixture of uranium-based materials in various forms.

By its decision of February, 05th 2014 in reference <3>, the American Competent Authority delivered the validation of the French approval certificate <2> expiring on June 30th 2018 for the package consisting in the TN-BGC 1 packaging loaded with the following contents of fissile material:

- Content n°11d, n°11e and n°11f: Non irradiated uranium-bearing materials in solid form;
- Content n°26: TRIGA fuel.

The validation was issued with additional special conditions to the French certificate. Among these, special condition 5.b prohibits the transportation of uranium metallic powder under content n°11.

By its letter in reference <5>, AREVA TN applied for a revision of the American Competent Authority validation in reference <3> to authorize the transport of non-pyrophoric uranium metallic powder. To date, no decision was issued yet.

By its letter in reference <4>, the U.S. Nuclear Regulatory Commission (NRC) staff, expressed its need for more information to determine if the TN-BGC-1 package meets the IAEA TS-R-1 requirements. You will find in Appendix 2 the answers to the questions raised in this letter.

Please note that in order to provide a complete answer, additional criticality analyses had to be performed. Calculation notes <6> and <7> provides a full description of these analyses. You will find attached an electronic version of these calculation notes.

These analyses show that, providing a reduction of uranium mass, IAEA requirements are met. The corresponding masses of uranium are listed in Appendix 1.

For your information, a transport of LEU from the Y12 facility (USA) to CERCA facility (France) is tentatively scheduled for the beginning of 2016.

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As a consequence, we hereby apply for the validation of the approval certificate <1> in America for the package consisting in the TN-BGC 1 packaging loaded with the following contents of fissile material with the mass limitations indicated in Appendix 1 :

- Content n°11 (except content n°11h): Non irradiated uranium-bearing materials in any solid form;
- Content n°26: TRIGA fuel.

We respectfully request the corresponding Competent Authority Certification to be issued no later than December 30th 2015.

Should you need any further information, please do not hesitate to contact us.

Gregory GALLAIS
Design Manager
Reactor Engineering Unit

Encl.: 2 CD-Roms, each containing :

- French Certificate of approval F/313/B(U)F-96 (Jbb) dated on October 10th, 2013;
- English translation of French Certificate of approval F/313/B(U)F-96 (Jbb) dated on October 10th, 2013;
- English translation of TN-BGC 1 Safety Analysis Report CEA DSN/STMR/LEPE/TNBGC1 DSEM 0600 Rev. 02 dated on October 11th, 2012;
- Criticality code validation for TN-BGC1- contents 11a, 11b, 11c and 11g, ref.TN-BGC1-0600 dated on June, 6th 2015;
- Criticality analysis for TN-BGC1- contents 11a, 11b, 11c and 11g, ref. TN-BGC1-0601 dated on June, 16th 2015;
- CEA technical note ref. SEC/T No. 89.18 dated on January 20th, 1989;
- Instruction of use (annexe 7) extracted from DSN STMR/LEPE TN-BGC1 NUT 0008 Ind.09 dated on October 13th, 2014;

Document which must not be made public :

- TN International technical note on the characterization of resin FS 69: ref. NTC-08-00118662-000 rev. 1 of June 22nd 2009;

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APPENDIX 1
Updated table of main characteristics for content No. 11 of TN-BGC 1

Transport of reprocessed uranium is prohibited.

The hydrogenous materials authorized for transport are PVC, polyurethane and polyethylene.

<u>For transport other than by air :</u>

The total mass of PVC, polyurethane and polyethylene (if authorized) is limited to 0,5 kg.

Content No.	Presence of hydrogen-bearing materials with higher hydrogen content than water authorised	Guaranteed containment diameter (mm)	Enrichment in ^{235}U	Maximum mass of U per package (kg)	Number of packages
11a	Yes	Ø ≤ 120	any	1,7	10
11b		Ø ≤ 100	any	15	5 [(1)]
11c		Ø ≤ 120	≤ 20 %	40	10
11d	No	100 < Ø ≤ 120	any	7	50
11e		60 < Ø ≤ 100	any	15	16
11f		Ø ≤ 60	any	40	50
11g		Ø ≤ 120	≤ 20 %	40	50

[(1)] Calculations show that N=10 is acceptable, However, for the sake of coherence with French certificate, the number of package is lowered to N=5 for content 11b.

<u>For transport by air :</u>

The maximum mass of ^{235}U per package is 7 kg.

The total mass of material with higher hydrogen content than water is limited to 0,4 kg.

The mass of water and the equivalent mass of other hydrogenous materials must not exceed 2 kg grams per package.

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APPENDIX 2
Answers to the Request for additional information for revalidation of model No. TN-BGC1 (French competent authority certificate of approval No. F/313/B(U)F-96 Rev. Jbb), Docket No. 71-3034, TAC No. L24858

Criticality Review

RAI 1

Identify the source of the reprocessed uranium; e.g., power reactor fuel assemblies, U-Mo fuel assemblies, as well as the anticipated fission products, uranium isotopes; e.g., uranium-234 and uranium-236, and plutonium isotopes in the content. In Section No. 3, "Criticality Study," of ANNEXE 11 of the application letter dated November 19, 2013, the applicant states that one of the intended contents is unirradiated reprocessed uranium. Paragraph 246 of TS-R-1 explicitly identifies the permissible quantities of plutonium, fission products, and uranium-236 which define uranium as unirradiated. The isotopic composition of reprocessed uranium can vary widely depending upon its source, and the TS-R-1 regulations for reprocessed uranium are very restrictive with regard to isotopes. Therefore, the applicant needs to provide specific information on both the source of the reprocessed uranium and the isotopic composition for this type of contents. The staff needs this information to proceed with its review to determine if the TN-BGC 1package with content 11 meets the requirements of para. 246 and 673 of TS-R-1 (2009 edition).

Since there is no transport need for reprocessed uranium, we propose to prohibit reprocessed uranium in the U.S. revalidation of content No. 11 in French Competent Authority Certificate No. F/313/B(U)F-96, Rev. Jbb.

RAI 2

Provide code benchmarking information and results for the computer codes, APLLO2 and MORET 4, and demonstrate that the TN BGC-1 package with content 11 meets the regulatory requirements of TS-R-1, para. 677, 678, 681 and 682. The applicant states in the safety analysis report that the APLLO2 and MORET 4 computer codes were used in the criticality safety analyses of the TN-BGC1 package. However, the applicant provided no benchmarking information for these codes. The applicant needs to provide information on the benchmarking of these codes, the resulting bias and bias uncertainties, corresponding corrections to the calculated k_{eff} values, and consequently the limit on the allowable ^{235}U contents and the Criticality Safety Index for each package. The staff needs this information to proceed with its review per the requirements of requirements of TS-R-1 (2009 edition), para. 677, 678, 681 and 682.

A criticality code validation has been performed using 251 critical experiments obtained from DICE. The corresponding calculation note in reference [1] is attached to this letter.
The critical experiments have been selected such that the geometric characteristics and material compositions are similar to that of TN-BGC1 package loaded with content No. 11. All the critical experiments were benchmarked with the use of the SCALE 6.0.
The results of the criticality benchmark were then used to derive the upper subcritical limit (USL) used in the additional calculations performed in the note in reference **Erreur ! Source du renvoi introuvable.** for TN-BGC 1 package loaded with content No. 11a, 11b, 11c and 11g.

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RAI 3

For content 11b:

1. Clarify if 15 kg rather 19.5 kg of uranium-235 (^{235}U) is the intended payload limit of content 11b;
2. Demonstrate criticality safety of the package that contains proposed hydrogenous materials;
3. Demonstrate that a 3.3 cm neutron poison resin layer would remain uniformly attached to the inner wall after the package thermal test of TS-R-1 para. 728(a);
4. Demonstrate criticality safety of the package assuming complete loss of neutron poison resin under hypothetical accident conditions (HAC); i.e., accident conditions of transport (ACT) if the assumed remaining 3.3 cm layer cannot be assured ;
5. Recalculate the Criticality Safety Index with $k_{eff} \leq 0.95$;
6. Correct Table 9 on page 23 of Chapter No. 8 to ensure that the payload quantities are consistent with the table presented in ANNEXE 11 of the application for contents 11c, 11f, and 11h.

The applicant requests authorization to transport 19.5 kg of ^{235}U for content 11b. The safety analysis report, however, analyzed criticality safety for a package containing 15 kg of ^{235}U for this content. The applicant needs to clarify the desired content limit and provide analyses to demonstrate the package meets the criticality safety requirements of TS-R-1 for that limit. The staff's own analyses show that k_{eff} for a TN-BGC1 package with 19.5 kg of ^{235}U will exceed 0.95 for a single package under normal conditions of transport as well as a package under HAC (ACT).

The applicant requests authorization of use of hydrogenous materials, such as polyurethane, polyethylene, and PVC as packaging materials. The safety analysis report, however, does not include an evaluation of the impact these types of materials have on criticality safety. The applicant needs to provide criticality safety analyses for packages that use hydrogenous materials, such as polyurethane foam and polyethylene sheet, as part of the packaging materials.

In its criticality safety analyses, the applicant assumed that there was a 3.3 cm neutron poison resin layer attached to the inner shell wall of the packaging after the package has endured a fire as described in TS-R-1 para. 728(a). The applicant needs to demonstrate this assumption is valid or redo the criticality safety analyses with an assumption that can be validated.

The staff also notes that a k_{eff} greater than 0.95 had been used as the acceptance criterion for criticality safety. This is not consistent with guidance in the 2008 edition of TS-G-1.1, para. VI.38. The applicant needs to demonstrate criticality safety for a single package as well as an array of packages under normal conditions of transport as defined in Para. 719 through 724 and HAC (ACT) as defined in Para. 726 through 733 of the 2009 edition of TS-R-1 using an acceptance criterion of $k_{eff} \leq 0.95$ per the guidance in the 2008 edition of TS-G-1.1, para. VI.38.

In addition, correct Table 9 in page 23 of Chapter No. 8 of the SAR to ensure that the payload limits are consistent with the table presented in ANNEXE 11 the of the application. There is no quantity limit in Table 9 for content 11c, 11f, and 11h.

The staff needs this information to proceed with its review to determine if the TN-BGC 1package with content 11 meets the requirements of para. 673 of TS-R-1 (2009 edition).

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1. Please note that the translation of S.A.R. contains inconsistencies. As a matter of fact, in para. 7.16. of Attachment 10 of Chapter 8, which deals with Content No. 11b, the table on page 69/131 should read as below (an additional line with air has been added to serve as reference) – for convenience, the corrections have been highlighted :

TN-BGC1 – 2N HAC (N=10) U (100 % ^{235}U) – Heterogeneous modelling Internal fittings: Ø = 100 mm Influence of water fog						
Water fog density	**Aluminium Spacer M (U) = 19,5 kg**		**Water spacer M (U) = 19,5 kg**		**Air spacer M (U) = 26,5 kg**	
	k_{eff}	k_{eff} + 3 s	k_{eff}	k_{eff} + 3 s	k_{eff}	k_{eff} + 3 s
Air	0,960	0,963	0,970	0,973	0,974	0,978
10^{-3}	0,953	0,957	0,902	0,905	0,967	0,970
10^{-2}	0,955	0,958	0,902	0,905	0,970	0,974
0,1	0,957	0,961	0,905	0,909	0,969	0,973
0,3	0,953	0,957	0,918	0,921	0,965	0,969
0,5	0,948	0,952	0,933	0,937	0,961	0,964
0,8	0,941	0,945	0,951	0,955	0,953	0,956
1,0	0,942	0,946	0,964	0,968	0,952	0,955

Thus, in the S.A.R, the intended payload limit of content No. 11b is actually 19,5 kg of uranium for any enrichment.
However, additional calculations have been performed (see note in reference **Erreur ! Source du renvoi introuvable.**) to demonstrate subcriticality of content No. 11b using an updated USL value as requested in RAI 2 of criticality review. The results show that the maximum mass of uranium is 15 kg for any enrichment, which corresponds to the limit proposed in Appendix 1.

2. Attachment 10 of Chapter 8 demonstrates criticality safety of the package loaded with various contents moderated with CH_2 which is more hydrogenated than proposed hydrogenous materials.

3. Thermal behavior of the FS 69 package resin was evaluated in the note in reference [3], attached to this letter. Please, note that this document is property of TN International and should not be made public.

The resin used in FS 69 package has the same chemical composition and physical properties than the resin used in TN-BGC 1 package. This study may then be extended to the TN-BGC 1 package resin.

The note in reference [3] describes fire test on three cylindrical samples of resin (diameter 240 mm, height 60 mm) placed in a 800°C flame for 30 minutes along with a very severe furnace fire test on a cylindrical sample of resin.

The main conclusions drawn from the direct flame test are the following:

- The resin kept its original form and remained solid during the whole thermal test (no melting);
- The crater created after scratching the burnt area has a depth which does not exceed 4 mm;
- The hydrogen content vary in the first 15 mm from the fired area;
- The total hydrogen loss in the resin is equivalent to a loss of 100% of hydrogen on a 9.7 mm thick layer.

The main conclusion derived from the furnace fire test is that the hydrogen loss in the resin is equivalent to a loss of 100% of hydrogen on a 14.25 mm thick layer.

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Considering these results, a conservative 15 mm reduction in the thickness of the resin was assumed in the safety criticality calculations which led to a remaining layer of 33 mm thick (considering the minimum thickness of 48 mm mentioned in para. 6.1 of chapter 2 of S.A.R.).

It should be emphasized that since the resin is casted into a 1.5 mm thick steel casing and regarding the results of drop tests described in Chapter 3 of S.A.R, the resin may not be lost after the thermal tests of TS-R-1 para. 728(a) and can be considered to remain uniformly attached to the inner wall of the casing.

4. A 33 mm neutron poison resin layer would indeed remain uniformly attached to the inner wall under hypothetical accident conditions (see precedent answer to RAI 3 - **3.**)

5. Additional analyses have been carried out for contents 11a, 11b, 11c and 11g with an updated K_{eff} based on a value of 0.9382 for USL (see note in reference **Erreur ! Source du renvoi introuvable.**). Theses analyses give the following results:

Content No.	Presence of hydrogen-bearing materials with higher hydrogen content than water authorised	Guaranteed containment diameter (mm)	Enrichment in ^{235}U	Maximum mass of U (kg)	Number of packages
11a	Yes	Ø ≤ 120	any	1,7	10
11b		Ø ≤ 100	any	15	10
11c		Ø ≤ 120	≤ 20 %	any	10
11g	No	Ø ≤ 120	≤ 20 %	40	50

6. There is no inconsistency in table 9 on page 23 of chapter 8. Masses in Chapter 8 of S.A.R. are defined from a criticality point of view whereas masses in the ANNEXE 11 of the application are limited by the maximum allowable mass of package.
Hence, to proceed with the review, the maximum allowable mass indicated in the French Competent Authority Certificate should be considered for contents No. 11c, 11f and 11h.
However, please note that content No. 11h is not requested in this application.

RAI 4

Provide the criticality safety analyses applicable to content 11f or identify its replacement. Section 4.11.1.4 of Chapter No. 8 of the SAR states CEA/SEC/T No. 89-18 dated 20 January 1989 demonstrates sub-criticality of the TN-BGC1 with content 11f. However, CEA/SEC/T No. 89-18 cannot be located within the submittal. The staff needs this information to proceed with its review per the requirements of para. 671 of TS-R-1 (2009 edition).

The document required has already been communicated by e-mail on 10th February 2014.
For convenience, it is also enclosed to this letter.

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Shielding Review

RAI 1

Justify why there is no shielding safety analysis if the content has significant quantities of gamma or neutron emitting isotopes (see above Criticality Review RAI 1). From the application, one of the intended contents is unirradiated reprocessed uranium. Since reprocessed fuel typically contains gamma and neutron emitting isotopes, the applicant needs to justify why there is no shielding safety analysis for the package if the unirradiated reprocessed uranium has significant quantities of gamma or neutron emitting isotopes. The staff needs this information to proceed with its review to determine if the TN-BGC 1package with content 11 meets the requirements of para. 569 and 657 of TS-R-1 (2009 edition).

As for Criticality Review RAI 1, we propose to prohibit reprocessed uranium in the U.S. revalidation of content No. 11 in French Competent Authority Certificate No. F/313/B(U)F-96 (Jbb).

Material Review

RAI 1

Clarify the maximum mass of PVC and polyurethane for non-air transport. A restrictive list of polymeric materials, namely polyurethane, polyethylene, and polyvinyl chloride (PVC), is specified in the French Certificate of Approval for sub-contents 11a, 11b, and 11c. For non-air transport, the certificate states that the maximum mass of polyethylene for sub-contents 11a, 11b, and 11c is limited to 500 g. However, no defined limitation for the mass of PVC or polyurethane in the package is specified. The staff needs this information to proceed with its review per para. 613,TS-R-1 (2009 Edition).

According to para. 613 of TS-R-1 (2009 edition), "The materials of the packaging and any components or structures shall be physically and chemically compatible with each other and with the radioactive contents. Account shall be taken of their behaviour under irradiation".

Chapter 9 of S.A.R. provides elements to demonstrate that risks associated to radiolysis and thermolysis of polymeric materials are taken into account.

Radiolysis analysis at HAC temperature for content No. 11 is covered by para.3.3.1 of Chapter 9 of S.A.R. The demonstration is made for an undefined mass of polymeric material but considering the radiolytic hydrogen generation rate of PVC as the highest of the authorized polymeric materials.

Thermolysis analysis at HAC temperature for content No. 11 is covered by para.3.3.2 of Chapter 9 of S.A.R. The calculation is performed for a maximum mass of polymeric material of 500 g.

Thus, for non-air transport, the total maximum mass of PVC, polyurethane and polyethylene should be limited to 500 g.

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RAI 2

Provide an analysis for the production of inflammable gases due to radiolysis and thermolysis of the polyethylene in the primary containment. The analysis should include a justification for the gas formation rates and the thermal decomposition thresholds considered.   A gas release analysis for uranium bearing contents is provided in Chapter 9 (§3.3.1) for a bounding content consisting of 45 kg of uranium enriched to 100% in ^{235}U, which exceeds the enrichment of contents 11a, 11b, and 11c. The applicant assumed a maximum polymer mass of 500 g and a maximum temperature of 144°C reached during HAC (ACT). However, the analysis uses a radiolytic efficiency for PVC which is much lower than that of polyethylene. The service temperature of polyethylene is also lower than that of PVC or polyurethane.   The staff needs this information to proceed with its review per para. 506, TS-R-1 (2009 Edition).

The document referenced NUREG/CR-6673 (2000) "Hydrogen Generation in TRU Waste Transportation Packages" by Lawrence Livermore National Laboratory, specifies the radiolytic efficiency for different materials at room temperature, among which, PVC and polyethylene.

Based on these values (see table D.11 of NUREG/CR-6673 (2000)), radiolytic efficiency of PVC and polyethylene may be determined at HAC temperature (144°C) as shown in the table below:

	PVC	Polyethylene
R ($J.mol^{-1}.K^{-1}$)	8,314	
HAC temperature T (K)	418	
Room temperature T_{0M} (K)	298	
Activation energy E_a ($J.mol^{-1}$)	33900	3340
Radiolytic G value at room temperature G_{Meff} (T_{0M}) (molecules/100 eV)	2,6	4,1
$G_{Meff}= G_{Meff}\,(T_{0M})^{*}\, e^{\frac{Ea\cdot(T-T_{0M})}{R\cdot T\cdot T_{0M}}}$	132,1	6,1

According to this table, gas formation rate of PVC is indeed lower than for polyethylene at room temperature. However, at HAC temperature (144°C), the PVC has a much higher radiolytic efficiency than polyethylene.

Thus, the results of gaseous release due to radiolysis derived using these radiolytic efficiencies cover the result which would be obtained with polyethylene radiolytic efficiency.

RAI 3

Clarify if the uranium-zirconium alloy is to be shipped as powder. If it is to be shipped as powder, provide details on the minimum particle size and additional controls to avoid flammability or pyrophoricity hazards. Provide details on inert gas purity, initial water content of powder, and procedures for loading and inerting the material.

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The French Certificate of Authorization states inerting conditions for the container, secondary containment, and the TN-BGC1 cavity, with a leakage rate below $1.33x10^{-5}$ Pa m^3s^{-1}. However, these conditions are insufficient to eliminate the potential for a pyrophoric reaction. Additional controls are required.

The staff needs this information to proceed with its review per para. 506, TS-R-1 (2009 Edition).

Uranium-Zirconium alloy is indeed to be shipped as powder inside a sealed primary container under an inert cover gas, as required by Chapter 9 of S.A.R.

Indeed, it is reminded that in para. 4 of chapter 9 and in Chapter 10 of S.A.R., contents which present a pyrophoricity hazard have to be transported under an inert gas atmosphere. This inert gas has to fill the package cavity, the secondary container cavity and the primary container cavity.

A detailed procedure for inerting the secondary container and the package cavity is given in the Appendix 7 of the Instructions of Use [5]. You will find this document attached.

This document specifies that prior to filling cavity with an inert gaz (N, He or Ar) at 1 bar, the cavity has to be placed under vacuum at a pressure of 1 mbar. This operation has to be repeated twice consecutively.

In addition, it has to be stressed that the pyrophoric nature of powders is defined in terms of ignition temperature. Many factors may influence this parameter; the most important is the specific area of powders.

The figure 5 of the review [4] (reproduced below) shows the variation of ignition temperature with specific area of uranium powders in oxygen. The ignition temperature of uranium powders varies quasi linearly from 320°C at low specific surface area (< 5 cm²/g) to 240°C for high specific surface area (> 50 cm²/g).

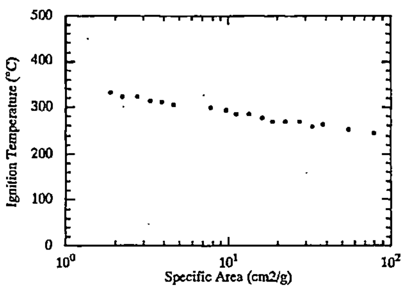

Fig. 5. Ignition temperatures of uranium powders in oxygen as a function of specific area (adapted from Tetenbaum et al., 1962).

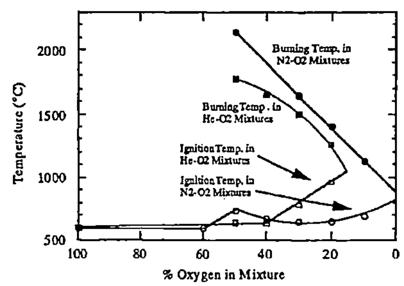

Fig. 6. The effect of oxygen concentration on ignition and burning temperatures of 8.5 mm uranium cubes (adapted from Schnizlein et al., 1959).

Figure 6 of the same review [4] (reproduced above) shows that ignition temperature tends to increase for lower oxygen rate. As a matter of fact, the ignition temperatures given in figure 5 for powders in oxygen will be lower than ignition temperatures for powders in an inert gas.

In a conservative manner, it is then considered that ignition temperature of uranium powders in an inert gas is not lower than 200°C.

In Chapter 5 of S.A.R, it is shown that in hypothetical accident conditions, the temperature of content would not exceed 144°C. Therefore the possibility of spontaneous ignition during transport is eliminated.

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In addition to this, a calculation of air penetration in cavity is performed in order to demonstrate that the quantity of air which enters the cavity during a transport would always be insufficient to generate a pyrophoric reaction.

It is reminded that for content No. 11, the pressure in cavity is set to atmospheric pressure for transport.

The main assumptions are the following:

- the cavity is assumed to be filled with an inert gas at room temperature and at a low ambient pressure (96 kPa);
- the ambient pressure considered for transport condition is maximized and corresponds to a high ambient pressure (104 kPa).

Considering that at some point the pressure of cavity would be in equilibrium with ambient pressure, making the leakage of air in the inerted cavity impossible, the total quantity of air entering the cavity is calculated as follow :

$$p_{air} = \frac{p_{atm\,maxi} - p_{atm\,mini}}{p_{atm\,maxi}} = 7{,}69\ \%$$

The corresponding volume of air penetrating the cavity is estimated based on the volumes provided in Table 2 of chapter 1 of S.A.R.:

- cavity internal volume is 37,6 L;
- TN90 external volume is 17,9 L.

The free volume is then 19,7 L.

As calculated above, the volume of air is 7,69% of the free volume, which is 1,52 L. The partial pressure of dioxygen in the air is taken to be 21% of $P_{atm\ maxi}$, that is : 21.8 kPa.

Assuming that the dioxygen in the air follow the ideal gas law, the mass of dioxygen in the cavity at T=298 K is :

$$m[O_2] = \frac{P.V}{R.T} \times 2.M[O_2] = \frac{2.18E^{+04} \times 19.7E^{-03}}{8.314 \times 298} \times 2 \times 16 = 0{,}43\ g$$

Considering ^{238}U, these 0,43 g of dioxygen can react with 3,16 g of uranium.

As a matter of fact, if such a reaction was to occur, i.e. if ignition temperature could be reached, the reaction would imply very little material and at some point would lack oxygen and self-extinguish.

To conclude, any pyrophoric reaction of uranium powders is prohibited for the following reasons :

- primary containers, secondary container (TN 90) and package cavity are under inert cover gas;
- the temperature range during transport is below the ignition temperature for uranium powders;
- the quantity of air which could enter the cavity is insufficient to produce a durable reaction.

AREVA

RAI 4

Clarify the maximum temperatures reached by the PVC and polyurethane in Content No. 11, and for polyurethane in Content No. 26. If the temperatures exceed 144°C, provide a valid gas generation analysis due to thermolysis/radiolysis of the covers.

The SAR analysis (Ch. 9, §3.3) assumes 144°C as the bounding temperature for the outside of the primary containment during HAC (ACT). A thermal evaluation was provided (Chapter No. 5, §7.23 and §8.3.3) assuming that PVC or polyurethane containers are used as covers for the primary container of an 80W content (Family 1 as defined in Chapter No. 5, §4). However, the SAR does not reference the covers being used for contents in Family 3 (i.e., zero power, Contents No. 11 and No. 26). In addition, the cover temperatures for a bounding Family 3 content are not provided.

The staff needs this information to proceed with its review per para. 506, TS-R-1 (2009 Edition).

Para. 8.4.1 of Chapter 5 of S.A.R. give the following results for zero power contents during HAC (Family 3):

- maximum temperature of gas in package cavity = 171°C;
- maximum temperature of gas in internal fitting cavity = 144°C.

Para. 8.3.1 of Chapter 5 of S.A.R. gives the following results for zero power contents during HAC (Family 3) :

- maximum temperature of internal fittings = 142°C;
- maximum temperature of content = 131 °C.

Hence, considering a maximum temperature of 144°C for polymeric material in zero power content is appropriate for thermolysis/radiolysis analyses of para. 3.3 of Chapter 9 of S.A.R.
Indeed, this consists in assuming that polymeric material reach the maximum temperature for gases in internal fitting cavity, which is penalizing.
Thus, the maximum temperature reached by the PVC and polyurethane in Content No. 11 and by polyurethane in Content No. 26 is 144°C.

In addition, for zero power contents, the maximum temperature raise of cavity gas may only occur during HAC. Indeed, since the content does not deliver any significant thermal power, the thermal power may only come from external conditions as defined in para. 728 of TS-R-1 (2009 edition).

Thermal power will then go from the outside of the package to the inside of the package.

As a consequence, taking into account the presence of covers around the content may cause a reduction of the maximum temperature of gases in internal fitting cavity due to the addition of the thermal inertia of covers.

AREVA

References:

[1] Criticality code validation for TN-BGC1- contents 11a, 11b, 11c and 11g, ref.TN-BGC1-0600 dated on June, 6th 2015;

[2] Criticality analysis for TN-BGC1- contents 11a, 11b, 11c and 11g, ref. TN-BGC1-0601 dated on June, 16th 2015;

[3] TN International technical note on the characterization of resin FS 69: ref. NTC-08-00118662-000 rev. 1 of June 22nd 2009;

[4] ANL report : "A review of the corrosion and pyrophoricity behavior of uranium and plutonium" by Terry C. Totemeier, ref: ANL/ED/95-2 of June 1995.

[5] Instruction for Use DSN STMR/LEPE TN-BGC1 NUT 0008 Version 09 dated on October 13th, 2014;

AREVA

Diffusion complémentaire

Diffusion interne (sans P.J.) :

TN INC.	G. MATHUES
DQS	C. VALLENTIN S. CARTIER L. MILET L. VIALLON E. BOUYER F. DARRAS T. CHAVALLARD F. FILIPPI (Chrono)
DT	A. FROMENT C. GRANDHOMME F. DERLOT J.-F. MALHAIRE B. KERR G. GALLAIS
BLFCS	R. LE BLEVENNEC L. STACHETTI

FRENCH REPUBLIC

AUTORITÉ DE SÛRETÉ NUCLÉAIRE

DEPARTMENT OF TRANSPORT AND SOURCES

APPROVAL CERTIFICATE FOR A PACKAGE DESIGN

F/313/B(U)F-96 (Jbb)

The French Governing Authority,

Pursuant to the request submitted by the Atomic Energy and Alternative Energies Commission (CEA) by letter CEA MR/DPSN/SSN/2012/177 dated 24 November 2012,

and in light of the safety analysis report CEA DSN/STMR/LEPE/TNBGC1 DSEM 0600 Ed. 02 dated 11 October 2012,

hereby certifies that the package design comprising the **TN-BGC 1** packaging described hereafter in appendix 0 index bb and

- loaded with:
  - uranium oxide powder, as described in appendix 2 index bb; or with
  - uranium metal ingots, as described in appendix 4 index bb; or with
  - uranium oxide fuel rods or fuel rod sections or pellets, as described in appendix 7 index bb; or with
  - solid uranium-bearing materials as described in appendix 11 index bb; or with
  - TRIGA fuel as described in appendix 26 index bb; or with
  - an aqueous solution of uranyl nitrate, as described in appendix 40 index bb; or with
  - U-Zr alloy fuel plates, as described in appendix 41 index bb; or with
  - uranium-bearing materials of diverse forms, as described in appendix 42 index bb;

is compliant, as a type B(U) package loaded with fissile materials, with the requirements of the regulations, agreements or recommendations listed below:

- International Atomic Energy Agency (IAEA) regulations for the safe transport of radioactive material, IAEA Safety Standards series, No. TS-R-1 2009 edition;
- European Agreement on the International Carriage of Dangerous Goods by Road (ADR);
- Technical Instructions for safe air transport of dangerous materials (ICAO-TI);
- Administrative decision of 29 May 2009 amended, on the carriage of dangerous goods by terrestrial routes (TMD decision);
- Instruction of 26 June 2008 pertaining to the technical rules and administrative procedures applicable to commercial air transport and regulation EC 859/2008 dated 20 August 2008 (EU OPS1).

Nonetheless, only contents no. 11 and no. 26 are authorised for air transport.

This certificate does not dispense the consignor from respecting the requirements established by the authorities of the countries across which or to which the package will be transported.

This certificate is valid until: **30th June 2018**

Record number: **CODEP-DTS-2013-054429**

Montrouge, 10th October 2013

SUMMARY OF CERTIFICATE REVISIONS

Produced	Expired	Type of revision and modifications	Authority	Certificate Ref. No.	Revision index							
					Body	t	0	1	2	3	4	5
		Reserved	DSND		Haj							
25/08/08	15/11/10	Prolongation: contents 2, 4, 7, 11 and 26	ASN	F/313/B(U)F-96	Iak	-	ak	-	ak	-	ak	-
25/08/08	31/08/13	Prolongation: contents 5, 6 and 15	ASN	F/313/B(M)F-96 T	Ial	al	al	-	-	-	-	al
	31/08/13	Reserved	DSND		Iam							
	31/08/13	Reserved	DSND		Ian							
12/02/09	31/08/13	Extension: inclusion of contents 1 and 3	ASN	F/313/B(M)F-96 T	Iao	ao	ao	ao	-	ao	-	-
10/04/09	31/08/13	Extension: modification of contents 1 and 3	ASN	F/313/B(M)F-96 T	Iap	ao	ao	ap	-	ap	-	-
04/11/09	31/08/13	Extension: modification of contents 1, 3, 5, 6 and 15; inclusion of contents 8, 9, 10, 18, 19, 20 and 23	ASN	F/313/B(M)F-96 T	Iaq	aq	aq	aq	-	aq	-	aq
	31/08/13	Reserved	DSND		Iar							
28/04/10	31/08/13	Extension: inclusion of content 39	ASN	F/313/B(M)F-96 T	Ias	as	as	-	-	-	-	-
04/06/10	31/08/13	Extension: cancels and replaces certificate F/313/B(U)F-96 (Iak)	ASN	F/313/B(U)F-96	Iat	-	at	-	at	-	at	-
	31/08/13	Reserved	DSND		Iau							
04/08/10	31/08/13	Extension: inclusion of content 42	ASN	F/313/B(U)F-96	Iav	-	av	-	-	-	-	-
	31/08/13	Reserved	DSND		Iaw							
10/11/10	31/08/13	Extension: inclusion of content 40	ASN	F/313/B(U)F-96	Iax	-	ax	-	-	-	-	-
10/05/11	31/08/13	Extension: modification of content 40	ASN	F/313/B(U)F-96	Iay	-	ay	-	-	-	-	-
17/08/11	31/08/13	Extension: inclusion of CH_2	ASN	F/313/B(U)F-96	Iaz	-	az	-	az	-	az	-
20/04/12	31/08/13	Extension: inclusion of content 46	ASN	F/313/B(M)F-96 T	Iba	ba	ba	-	-	-	-	-
		Prolongation: contents 2, 4, 7, 11, 26, 40, 41 and 42	ASN	F/313/B(U)F-96	Jbb	-	bb	-	bb		bb	-
10/10/13	30/06/18	Prolongation : contents 2, 4, 7, 11, 26, 40, 41 et 42	ASN	F/313/B(U)F-96	Jbb	-	bb	-	bb		bb	-
		Reserved	DSND		Jea							

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Iaz	-	az	-	-	-	az	-	-	-	-	-	-	-	-	-	-	-	-	-	-	az	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Iba	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	ba
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ANNEXE 0

TN-BGC 1 PACKAGING

1. PACKAGING DEFINITION

The packaging is designed, manufactured, inspected, tested, maintained and used in compliance with Safety Analysis Report CEA DSN/STMR/LEPE/TNBGC1 DSEM 0600 Rev. 02 dated 11 October 2012.

The packaging consists of a parallelepiped cage inside which a generally cylindrical body equipped with a closure system and a shock absorber cover is fixed. The packaging is presented in figure 0.1.

The packaging design drawings are as follows:

- concept drawing - overall : TN 9990-65 (C);
- cage : TN 9990-118 (B);
- fitted plug : TN 9990-117 (B);
- shock absorber cover : EMB TNBGC PBC PDC CA 010001 A.

The main dimensions of the packaging are as follows:

- cage cross-section : 600 x 600 mm²;
- overall height of cage : 1 821 mm;
- diameter of body in centre section : 295 mm;
- diameter of cover : 466 mm;
- overall body length with cover fitted: 1 808 mm.

The maximum admissible mass of the loaded packaging for transport is 396 kg; its mass when empty is 280 kg.

In light of the tolerances on the dimensions and densities of the balsa and poplar wood in the packaging (shock absorber cover and base of body), the total mass of water in these elements is below 1670 grammes.

The packaging comprises the principal sub-assemblies described below.

1.1 Cage

The cage is a structure made of 30 x 30 mm, 2 mm thick aluminium tubes.

Reinforced passages are built in at two heights to enable introduction of the forks of a forklift truck, in order to handle the packaging.

Frames are provided inside the cage to connect the cage to the body. They are welded to the vertical struts of the cage and drilled to allow the passage of the body mounting bolts.

1.2 Body

The cavity has a useful diameter of 178 mm and useful length of 1,475 mm. It is formed from a 6 mm stainless steel shell (that provides most of the radial gamma shielding) and an 8 mm base also made of stainless steel.

A second 1.5 mm stainless steel shell with an internal diameter of 292 mm cooperates with the first shell to delimit a space filled with resin (minimum thickness 48 mm), which acts as a neutron absorber and an active thermal insulation.

From the inside of the packaging towards the outside, the base also features a 25 mm thick steel diffuser plate made of high-strength steel, a 24 mm layer of resin, a false bottom, a wooden shock-absorbing disk and a stainless steel plate.

In the upper part, a stainless steel machined flange is welded to the two shells to receive the closure system described below.

1.3 Closure system

The body cavity is closed using a system composed of 3 main parts: a plug, a compression ring and a bayonet ring.

The plug is held against the body by the compression ring. This component is screwed into the bayonet ring, which itself is compressed against the body flange.

In the centre of the plug, a hole fitted with a quick-connect coupling allows the package to be de-pressurised before dispatch and re-pressurised to atmospheric pressure upon arrival before unloading. This hole is closed with a cap.

The leaktight seal between the plug and body and the quick-connect cap is formed by two pairs of O-ring seals. The spaces between the seals both communicate with the same inspection port used to verify the leaktight properties of the closure system.

The two O-ring seals defining the limit of the containment system are made of THT silicone, with a hardness rating of shore 65. These seals are numbered 11 and 13 on the TN 9990-65 (C) plan. The other two seals (12 and 14) are Viton seals.

1.4 Shock absorption system

A leaktight shock absorber cover is placed over the top of the body and the closure system.

It is composed of two steel plate compartments, that closer to the body is filled with resin, the other is filled with wood.

The cover is attached to the body using two bent rods that fit into the lugs on the body, and with two clamping pins, the ends of which are bolted onto the cover and welded to the packaging body.

1.5 Handling and tie-down components

The cage is used to handle and secure the packaging.

The packaging can be handled either in a horizontal or a vertical position.

The packaging is transported in horizontal or vertical position, according to the principles set out in the instructions for use of the package model and in the safety file of the package model:

- in horizontal position: the packaging is stowed on the floor and tied down around the cage by strapping. Only one stacking level is allowed. Wooden structures can be used between two packagings and around the cages.

- in vertical position: the packagings are grouped in batches and held by straps which are arranged above the cage and at mid-height. The leg is stowed. Stowing devices (for example corner beams) are added in the upper section. The packagings can be grouped in rows of 2.

Stowing and tie-down must be performed on the basis of a pre-set procedure checked according to the provisions of the quality management system.

1.6 Safety functions and elements important for safety

The main safety functions and elements important to safety are:

- the **containment** function offered by the packaging containment system, represented by the inner shell, the base of the body and their circumferential welds, the plug and the quick-connect cap, which are fitted with silicone internal seals;
- **radiological protection**, mainly provided by:
  - lateral shielding represented by the stainless steel of the inner shell and the outer shell for the main part of the shielding against gamma radiation, and by the resin (48 mm minimum) for the shielding against neutron radiation;
  - base shielding represented by the stainless steel of the base of the cavity and the two closure plates as well as the carbon steel in the distribution baffle for the main part of the shielding against gamma radiation, and by the resin (24 mm minimum) and wood for the shielding against neutron radiation;
  - top shielding represented by the stainless steel of the plug and the sheet metal cover for the gamma protection, and by the resin (min. 24 mm) and wood contained in the cover for the neutron protection;
- **criticality safety** provided by the confinement system which comprises the elements described in the content appendix and:
  - the packaging: geometry (maximum diameter of the packaging to favour interactions, cage), the materials used, the composition and thickness of the neutron-absorbing resin (hydrogen and boron content, thickness of burnt resin);
  - the internal fittings: geometry of shims, constituent materials of shims (aluminium), geometry (diameter, thickness) of container, material used for container;

For the internal fittings, the parameters important to safety are indicated in the table below.

Internal fitting	Internal diameter (mm)	Thickness (mm)	Material
TN90	≤ 120	≥ 2	Z2 CN 18-10
AA-41 - AA203 - AA204	≤ 115	≥ 2	Z2 CN 18-10
TN90 type 2	≤ 130	$4 \leq e \leq 5$	Z2 CN 18-10
E7	≤ 60	≥ 2	UA4G

- the **dissipation of internal thermal power** via radiation between the radioactive materials and the inner shell within the body, by conduction in the body and heat exchange between the body and the ambient air;
- **impact protection**, provided by the shock absorber cover and the cage;
- **fire protection**, provided by the radiological protection and the cover. The body and the cover are fitted with fuse plugs that prevent the risk of overpressure due to the accumulation of steam.

2. MEASURES TO BE TAKEN BY THE CONSIGNOR PRIOR TO SHIPPING THE PACKAGE

The packaging must be used according to procedures that comply with the instructions for use in chapter 10 of the safety analysis report CEA DSN/STMR/EMBAL/LEPE/DSEM 0610 Rev. 02 dated 11 October 2012.
Furthermore, for the leakage test of the closure system via the inspection port, the leakage rate must remain below 6.65 x 10^{-4} $Pa.m^3.s^{-1}$ SLR.

3. MAINTENANCE PROGRAMME

The packaging must be used according to procedures that comply with the instructions for use in chapter 10 of the safety analysis report CEA DSN/STMR/EMBAL/LEPE/DSEM 0610 Rev. 02 dated 11 October 2012.

4. NOTIFICATION AND REGISTRATION OF SERIAL NUMBERS

The applicable authorities must be kept informed of any packaging that is taken out of service or transferred to another owner. With this in mind, it is the ceding owner who is responsible for providing the name of the acquiring owner.

5. QUALITY ASSURANCE

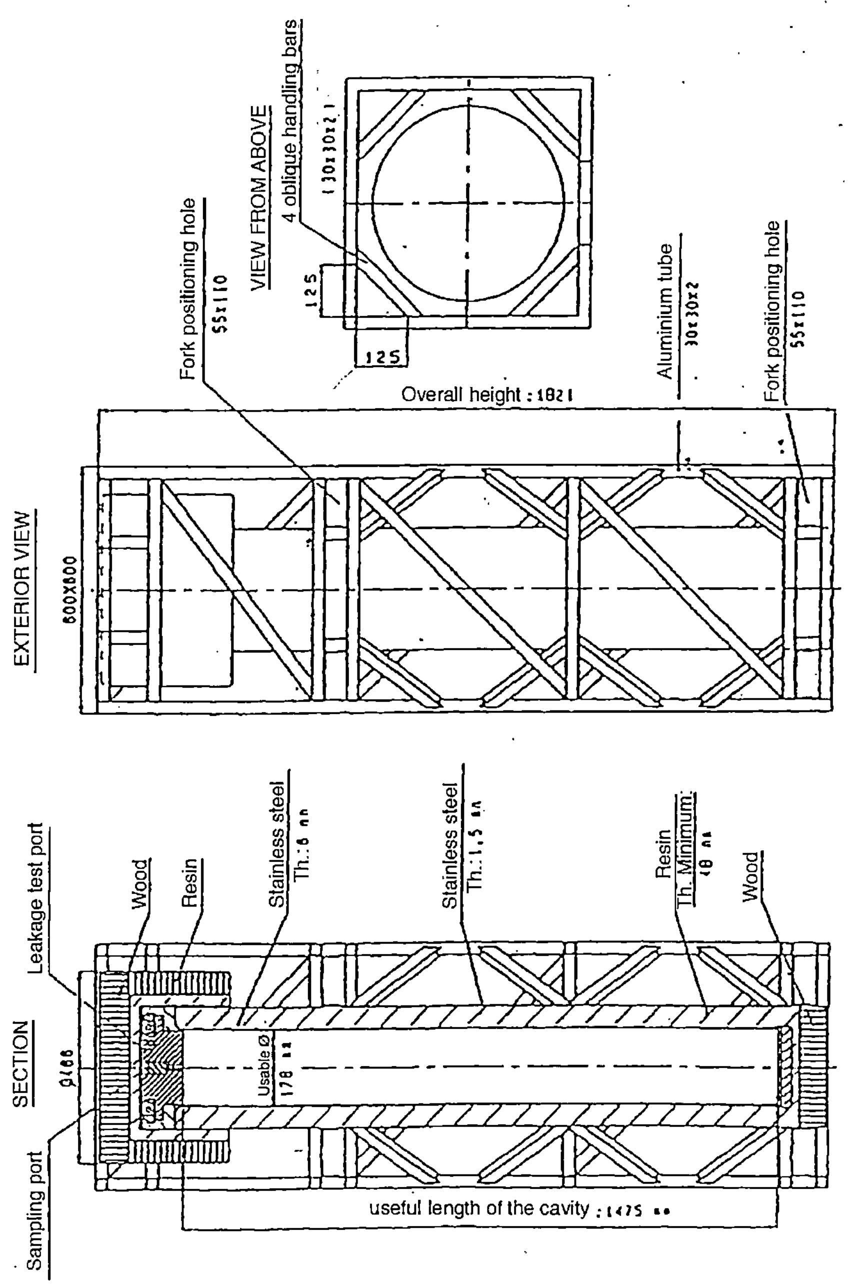
The quality assurance principles applied during the design, manufacturing, inspection, testing, maintenance and use of the package must comply with those described in chapter 11 of the safety analysis report CEA DSN/STMR/EMBAL/LEPE/DSEM 0611 Rev. 01 dated 17 July 2012.

6. ADDITIONAL REQUIREMENTS IN THE EVENT OF CONFINED TRANSPORT

If packages are shipped inside a type CB9 transport crate, the heat dissipation conditions may be modified. The thermal power must therefore be below 4W per package and 48W for all packages transported.

FIGURE 0.1

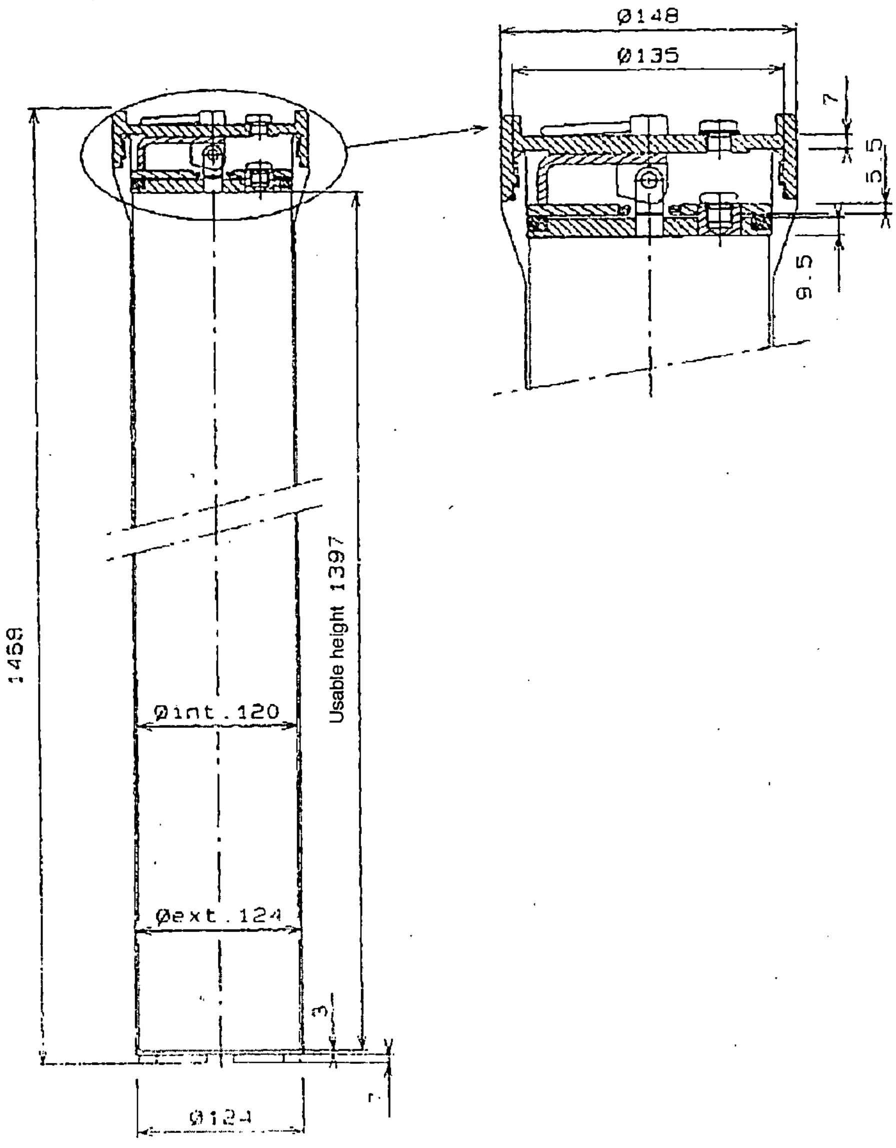
SCHEMATIC VIEW OF PACKAGING



Note: dimensions are in millimetres.

FIGURE 0.2

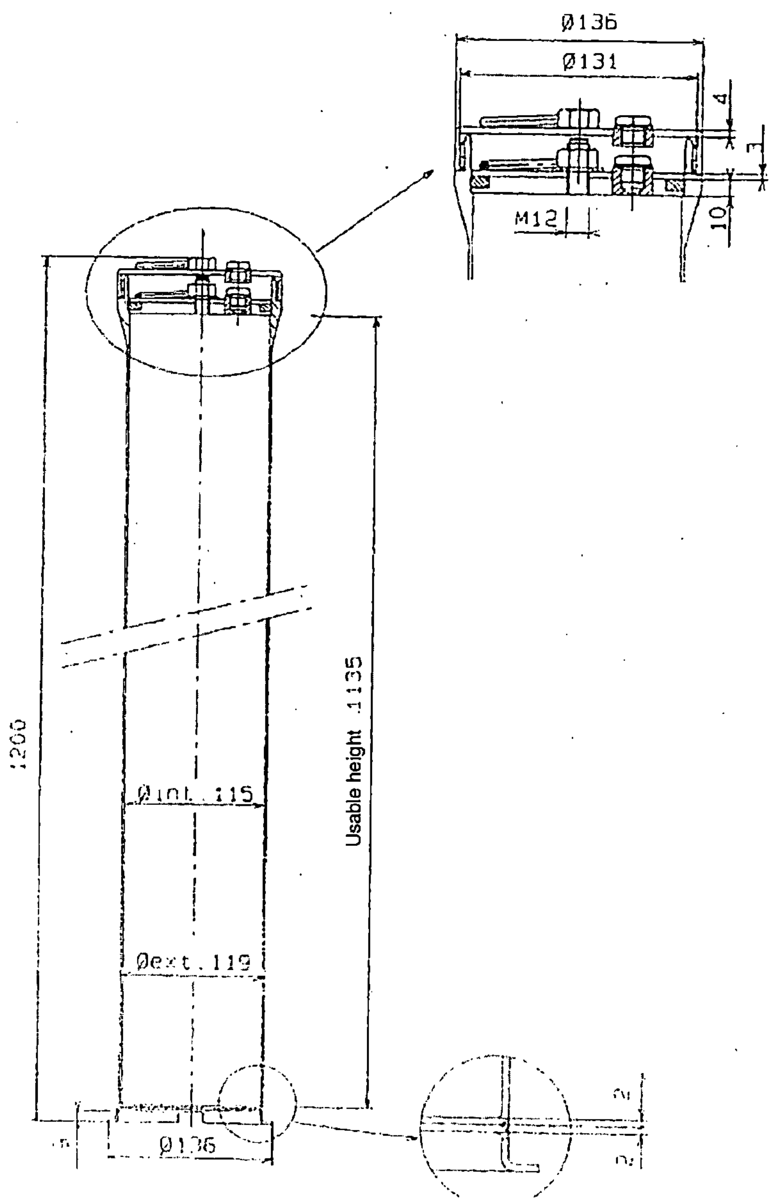
SCHEMATIC VIEW OF TN 90 CONTAINER



Note: dimensions are in millimetres.

FIGURE 0.3

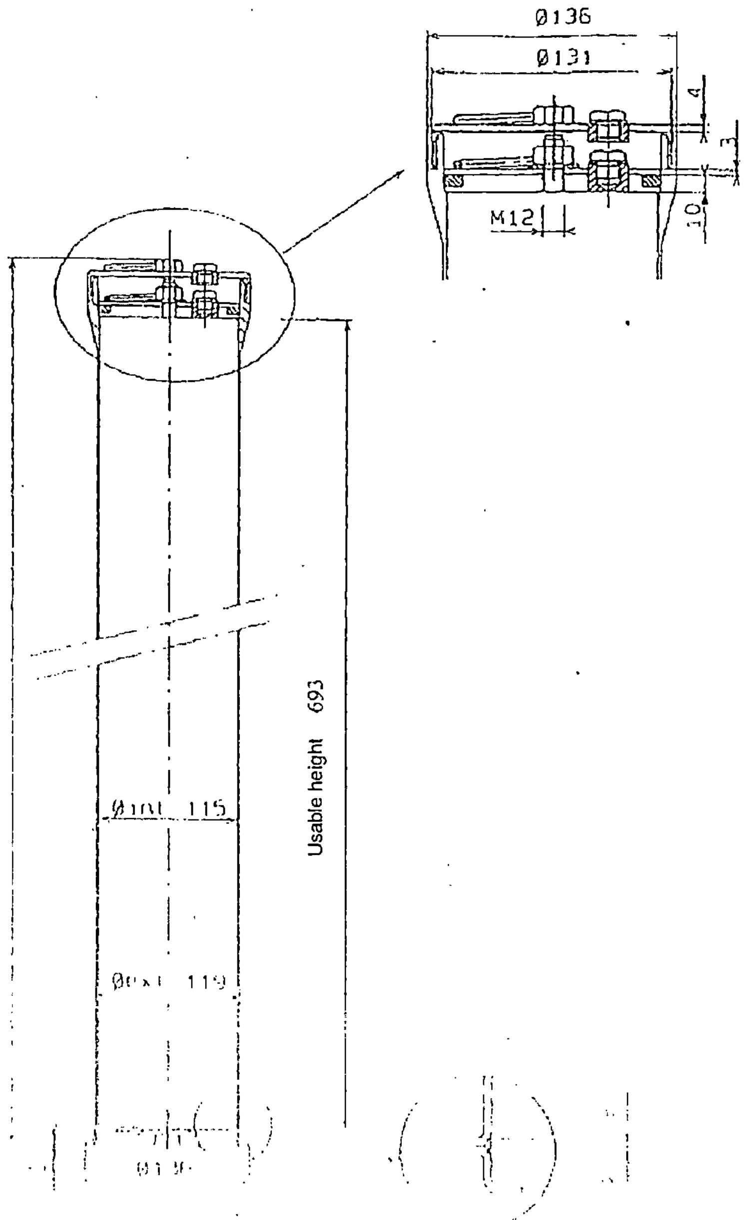
SCHEMATIC VIEW OF AA 204 CONTAINER



Note: dimensions are in millimetres.

FIGURE 0.4

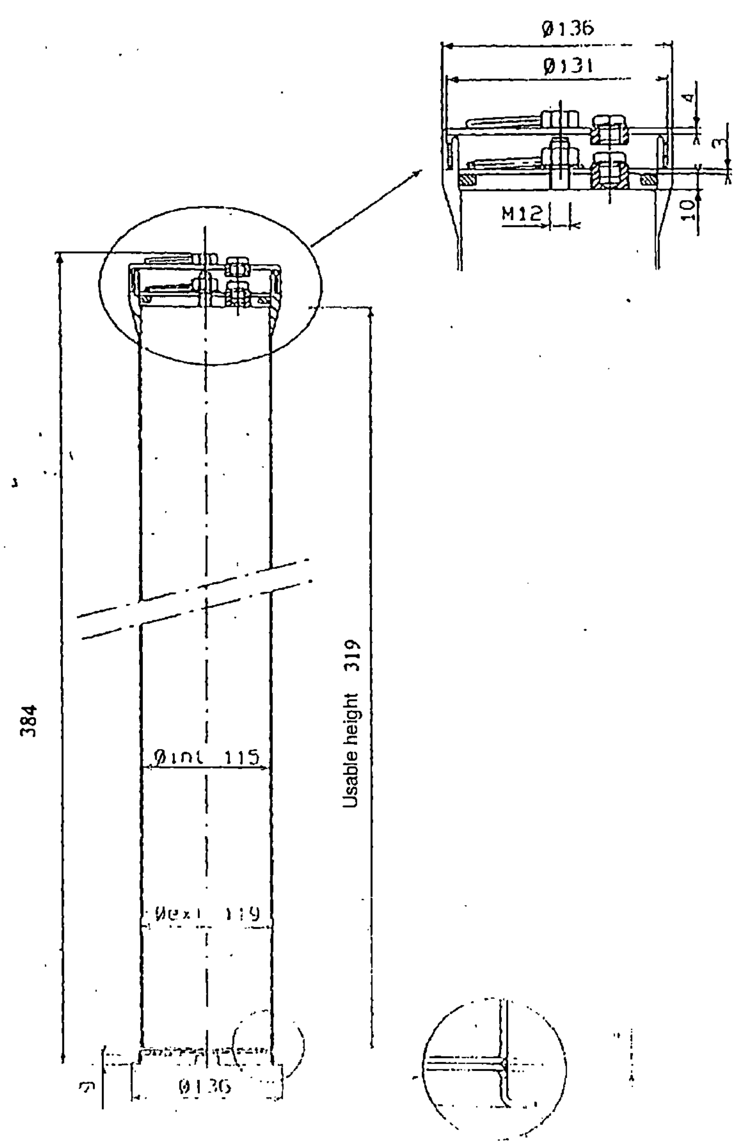
SCHEMATIC VIEW OF AA 203 CONTAINER



Note: dimensions are in millimetres.

FIGURE 0.5

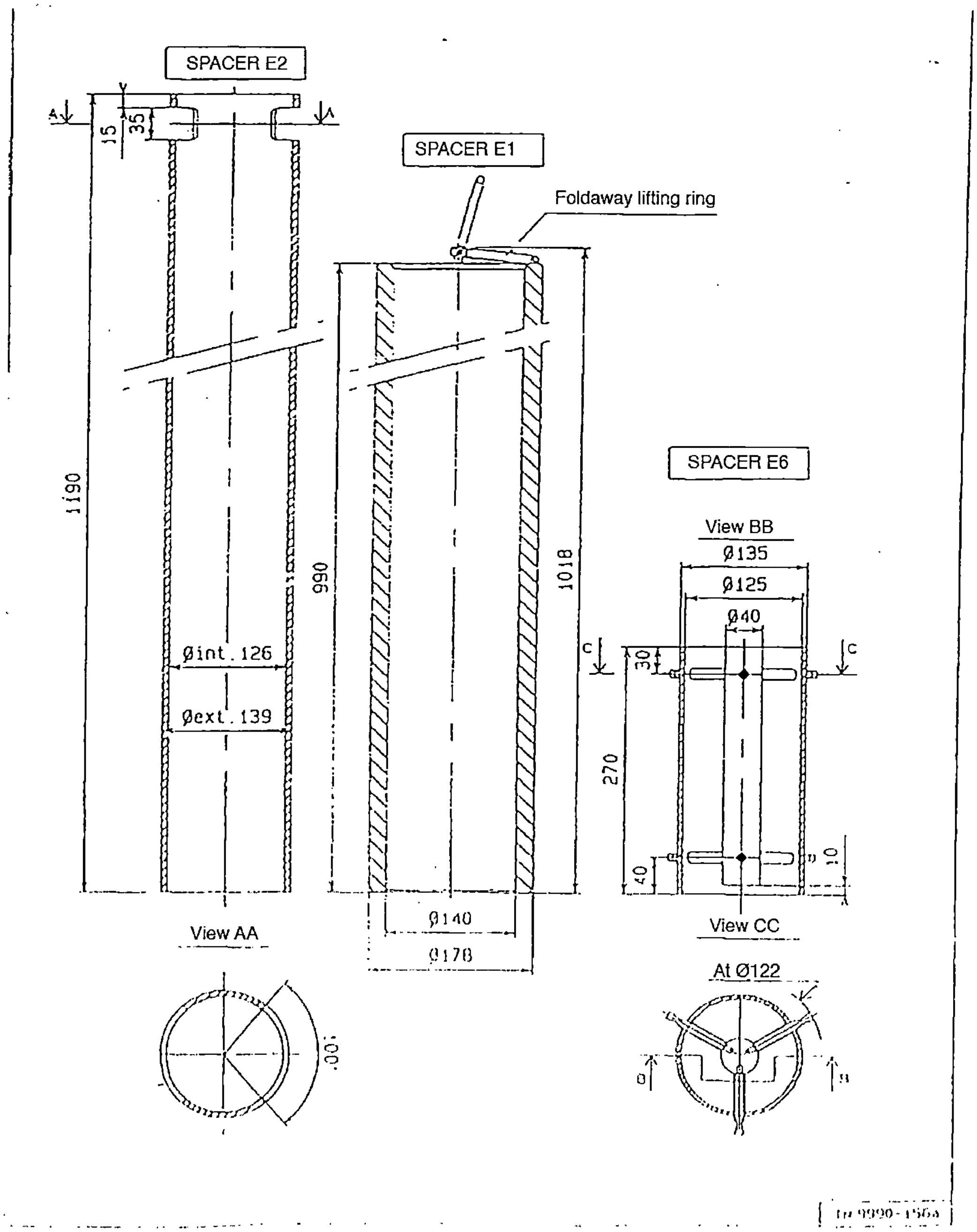
SCHEMATIC VIEW OF AA 41 CONTAINER



Note: dimensions are in millimetres.

FIGURE 0.6

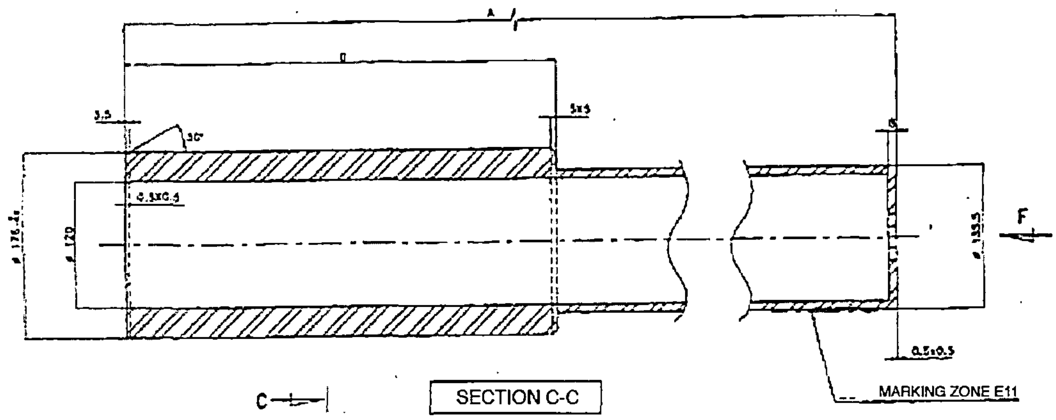
SCHEMATIC VIEW OF SPACERS E1, E2, E6

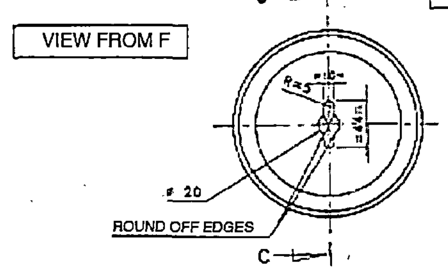


Note: dimensions are in millimetres.

FIGURE 0.7

SCHEMATIC VIEW OF SPACERS E8, E10, E11, E12





PWR	A	B
E8	707	407
E10	265	165
E11	1081	407
E12	677	407

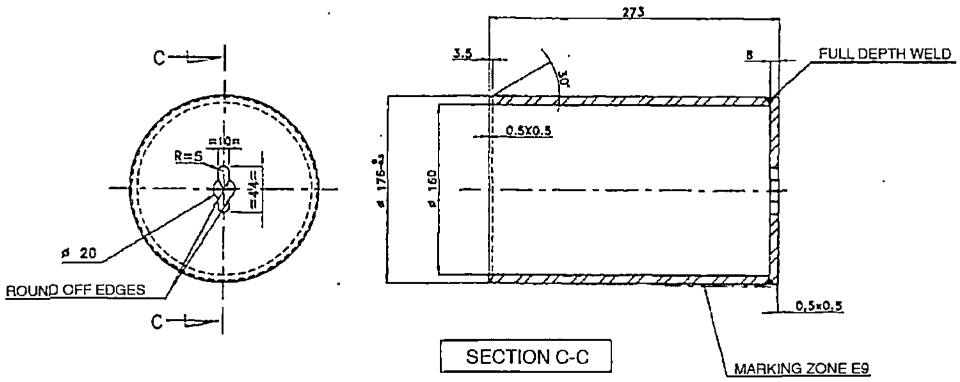
VERSIONS
AA 203
AA 204
AA 41X1
AA 42X2

SUMMARY TABLE SHOWING SHIMS

Note: dimensions are in millimetres.

FIGURE 0.8

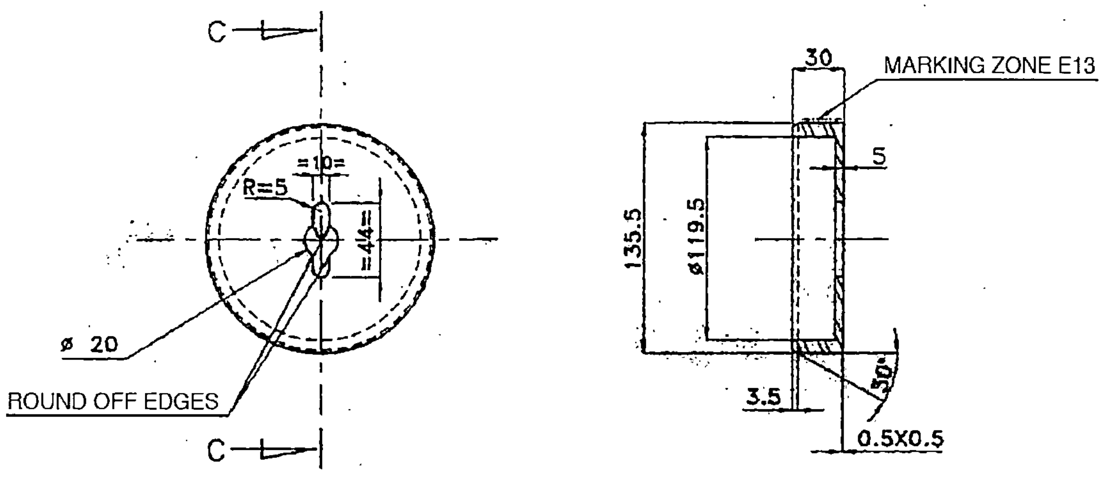
SCHEMATIC VIEW OF SPACER E9



Note: dimensions are in millimetres.

FIGURE 0.9

SCHEMATIC VIEW OF E13 SPACER FOR AA 41



Note: dimensions are in millimetres.

ANNEXE 11

CONTENT NO.11

NON-IRRADIATED SOLID URANIUM-BEARING MATERIAL

1. DEFINITION OF AUTHORISED CONTENT

This content comprises uranium-based solids. The presence of traces of plutonium, in the order of grammes, is permitted.

The uranium is non-irradiated. In the event that the uranium is reprocessed, the material should not have been irradiated at any moment post-reprocessing.

The presence of other materials than those defined in this appendix (content and internal fittings) is excluded.

__Isotopic composition and masses__

The Uranium 235 enrichment level is unimportant, but authorisation is granted for a single type of uranium material per package (single isotopic composition).

For transport by air: the maximum quantity of uranium 235 transported in a TN-BGC 1 package is 7 kg. In this case a quantity of 400 g of materials with a greater hydrogen content than water is authorised.

For transport other than by air: the maximum admissible masses are specified in the following table:

Content No.	Presence of hydrogen-bearing materials with higher hydrogen content than water authorised	Guaranteed containment diameter (mm)	Enrichment in ^{235}U	Maximum mass of U (kg)	Number of packages
11a	Yes	$\varnothing \leq 120$	any	2	10
11b		$\varnothing \leq 100$	any	19,5	5
11c		$\varnothing \leq 120$	≤ 20 %	40	10
11d	No	$100 < \varnothing \leq 120$	any	7	50
11e		$60 < \varnothing \leq 100$	any	15	16
11f		$\varnothing \leq 60$	any	40	50
11g		$\varnothing \leq 120$	≤ 20 %	40	50
11h		$\varnothing \leq 115$	≤ 30 %	40	25

The hydrogen-bearing materials authorised for transport are polyethylene, polyurethane and PVC. The presence of polyethylene in sub-contents 11d, 11e, 11f, 11g and 11h is not permitted. In other sub-contents, its maximum mass is limited to 500g.

In the event it proves impossible to guarantee a single isotopic composition per package, the mass limitations are as follows:

- If the uranium is enriched to more than 20 %, the maximum transportable mass of uranium is 7 kg in the absence of materials with a higher hydrogen content than water and 2 kg in the presence of such materials;

- If the various uranium products are enriched to less than or equal to 20 % the maximum transportable mass of uranium is 40 kg in the absence or presence of materials with a higher hydrogen content than water.

Physical characteristics

Any density

Chemical form

The material is exclusively in one of the following chemical forms (or in the form of a mixture of these forms):

- Metallic Uranium,
- Uranium oxides: UO_2, UO_3, U_3O_8,
- Uranium tetrafluoride: UF_4
- Uranium nitrides: UN, U_2N_3, UN_2,
- Uranium carbides: UC, UC_2 and U_2C_3,
- Uranium alloyed with the following metals: aluminium (Al), Molybdenum (Mo), Silicon (Si), Zirconium (Zr).

Special form

The material is not in a special form.

Specific

The activity of the content must be such that, given the nature and energy of the radiation emitted, the regulatory limits for dose-rates around the package are not exceeded.

2. INTERNAL FITTINGS AND CONTAINERS

The metallic uranium powder is placed in a primary container, which may be metal boxes, flasks or polymer casings. The whole assembly is then placed within a container representing the secondary containment.

The secondary containers must have a maximum internal diameter of 120 mm and a thickness of 2 mm. They may be TN 90, AA 204, AA 203 or AA 41 containers (see figures 0.2 to 0.5).

In the event that the required internal diameter is less than 120 mm, the secondary container used must be a TN 90 type; the positioning and radial securing inside the TN 90 container must feature E7 type spacers.

The following spacers (figures 0.6 to 0.9) should be used to position and secure the secondary container in the packaging cavity:

- for the TN 90: Spacers E1 and E2,
- for the AA 204: Spacers E1 and E10 or E6,
- for the AA 203: Spacers E1 and E8,
- for 1 x AA 41: Spacers E1 and E11,
- for 2 x AA 41: Spacers E1, E12 and E13,
- for 3 x AA 41: Spacers E1, E9 and 2 × E13.

The total mass of the loaded internal fittings (AA41, AA203, AA204 and TN90 (materials + primary containment) must not exceed 60 kg.

The maximum admissible mass of the whole load within the cavity of the TN-BGC 1 package (material transported, primary/secondary containments & shims) is 116 kg.

Special provisions

When the content is in powder form and if the primary container has been in storage for over a month, the crown must be renewed before loading into the secondary container then into the packaging.

The time between the closure of the secondary containment in the dispatching plant and the arrival of the package at the destination must be less than one year.

3. CRITICALITY STUDY

This is covered by attachments 1, 10 and 12 in chapter 8 reference CEA DSN/STMR/LEPE/TNBGC1 DSEM 0608 Rev. 01 dated 17 July 2012 and note CEA/SEC/T n°89-18 dated 20 January 1989.

Their sub-criticality related characteristics are listed below:

For contents 11a, 11b and 11c, it permits the presence of **hydrogenated materials with a hydrogen concentration greater than that of water** and/or water penetration in all the unoccupied space in the packaging, including in the containment system.

For contents 11d, 11e, 11f and 11g, it permits the presence of **hydrogenated materials with a hydrogen concentration less than or equal to that of water** and/or water penetration in all the unoccupied space in the packaging, including in the containment system.

Criticality Safety Index:
For contents 11a and 11c: CSI = 5 (number "N" : 10).
For content 11b: CSI = 10 (number "N" : 5).
For content 11e: CSI = 3.125 (number "N" : 16).
For contents 11d, 11f and 11g: CSI = 1 (number "N" : 50).
For content 11h : CSI = 2 (number "N" : 25)

4. SPECIAL PROVISIONS

When the material is in a metallic form, particular attention should be paid to the surface condition of the content when loading into their packaging. The surface of the content should be free of scoring or hydrides.

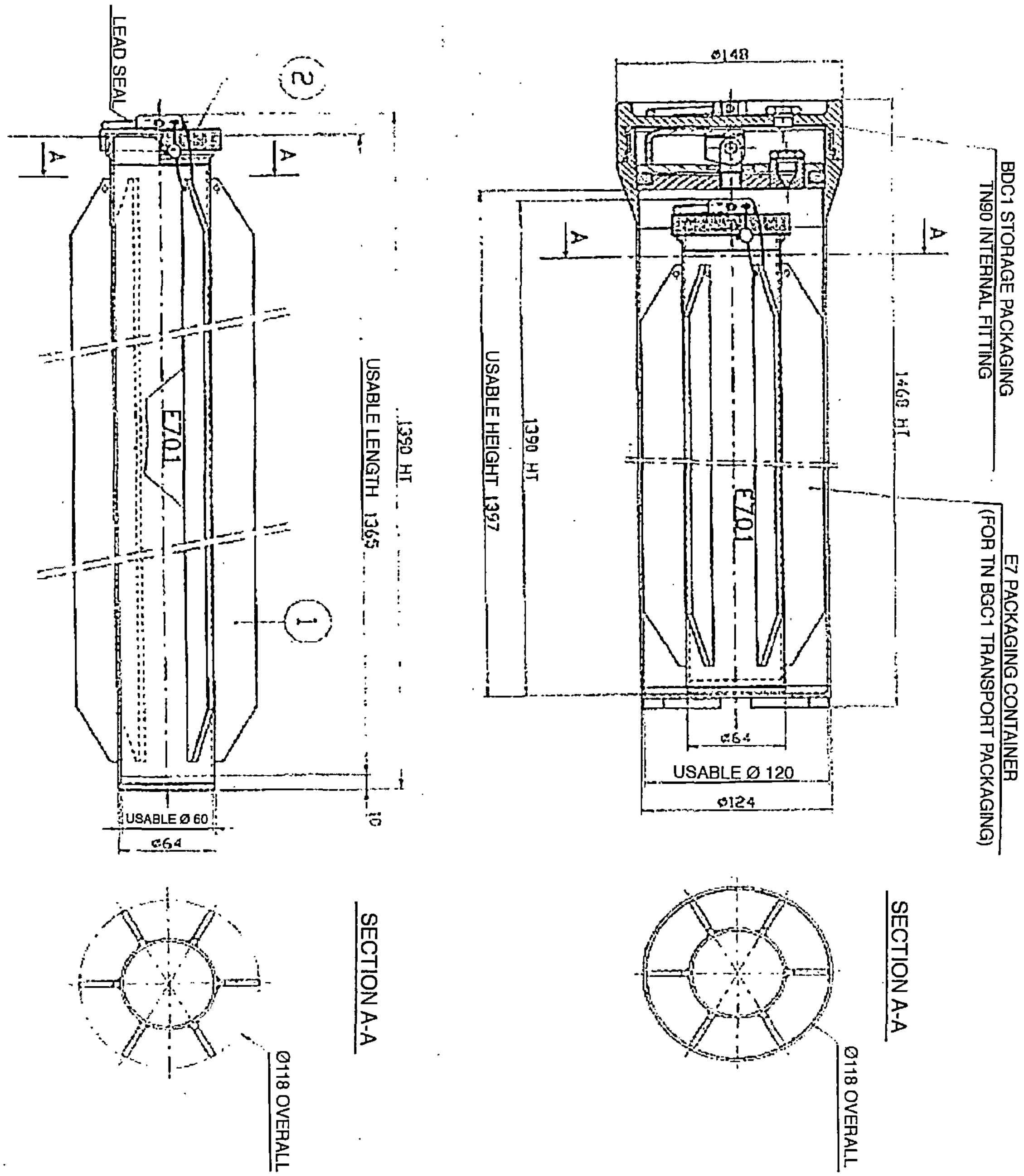
In the event that metallic powders are transported, the containers, the secondary containment and the TN-BGC1 cavity itself must all be inerted at ambient pressure, and a leak test must be carried out on the secondary container (leakage rate below 1.33×10^{-5} $Pa.m^3s^{-1}$).

FIGURE 11.1

SCHEMATIC VIEW OF E7 TYPE CONTAINER

AND

PACKING OF URANIUM-BEARING MATERIALS



Note: dimensions are in millimetres.

ANNEXE 26

CONTENT NO. 26

TRIGA FUEL

1. DEFINITION OF AUTHORISED CONTENT

This content comprises non-irradiated bars of TRIGA fuel elements.

These bars are based upon U ZrHx (where x is between 0 and 2); they are in cylindrical form and are of one of two types- standard or thin, with the following geometric characteristics:

- Standard: diameter = 3.63 cm; length = 12.7 cm,
- Thin: diameter = 1.29 cm; length = 18.6 cm.

The uranium does not come from reprocessing.

The standard bars are drilled, hydrogenated by the centre, the diameter of the hole is 6.35 mm.

The schematic view of standard and thin TRIGA fuel elements is presented in figure 26.1.

The hydrogen-bearing materials authorised for transport are polyethylene and polyurethane. The presence of hydrogen-bearing materials with a hydrogen concentration greater than that of water is not permitted.

The content must be free of all traces of humidity.

The presence of materials other than those defined in the approval certificate is excluded.

Isotopic composition and masses

The maximum 235U enrichment level is 20%. The mass content of U varies between 8 and 47% depending on the type of element:

TYPE	U (% by mass)	ZrH_x (% by mass)	U-Zr (g/cm^3)	U-ZrH_2 (g/cm^3)
Composition of standard TRIGA fuel elements				
103	8	92	6,9	6,04
105	12	88	7,1	6,22
107	12	88	7,1	6,22
117	21	79	7,4	6,64
119	31	69	8,1	7,24
Composition of thin TRIGA fuel elements				
424	47	53	9,3	8,40

Maximum transportable quantities

The maximum transportable quantities are given in the tables below.

- For **transport by air**: the maximum mass of uranium transportable in a TN-BGC 1 package is dependent on the type of fuel element as shown by the table below:

TYPE	Total mass of U (kg)
103	1,1
105	1,7
107	1,7
117	3,3
119	5,3
424	6,6

- For **other modes of transport**: the maximum mass of uranium transportable in a TN-BGC 1 package is dependent on the type of fuel element as shown by the table below, however, the maximum masses defined in Paragraph 2 for the loads applicable to internal fittings and packagings must be respected:

TYPE	Total mass of U (kg)
103	9
105	14
107	14
117	27
119	43
424	76

Special form

The material is not in a special form.

Specific

The activity of the content must be such that, given the nature and energy of the radiation emitted, the regulatory limits for dose-rates around the package are not exceeded.

Special provisions

For transport by air, the mass of water present with the fissile materials, independently of the hydrogen-bearing materials of the package, is less than 1,200 g, or 1,950 g, depending on whether they are standard or thin fuel elements.

The time between the closure of the secondary containment in the dispatching plant and the arrival of the package at the destination must be less than one year.

2. INTERNAL FITTINGS AND CONTAINERS

The TRIGA bars are placed inside cardboard protective tubes, which in turn are placed in a secondary container.

The secondary containers that may be used must have a maximum internal diameter of 120 mm and a thickness of 2 mm. They may be TN 90, AA 204, AA 203 or AA 41 containers (see figures 0.2 to 0.5).

A primary container (Type E7) (see figure 26.2) can be used with the TN 90 for uranium-bearing materials.

The following spacers (figures 0.6 to 0.9) should be used to position and secure the container in the packaging cavity:

- for the TN 90 : Spacers E1 and E2;
- for the AA 203 : Spacers E1 and E8;
- for the AA 204 : Spacers E1 and E10;
- for 1 x AA 41 : Spacers E1 and E11;
- for 2 x AA 41 : Spacers E1, E12 and E13;
- for 3 x AA 41 : Spacers E1, E9 and 2 x E13.

The total mass of the loaded internal fittings (AA41, AA203, AA204 and TN90 (materials + primary containment) must not exceed 60 kg.

The maximum admissible mass of the whole load within the cavity of the TN-BGC 1 package (material transported, primary/secondary containments & shims) is 116 kg.

3. CRITICALITY STUDY

This is covered by attachments 9 and 11 in chapter 8 reference CEA DSN/STMR/LEPE/TNBGC1 DSEM 0608 Rev. 01 dated 17 July 2012.

It permits the presence of hydrogenated materials with a hydrogen concentration less than or equal to that of water and/or water penetration in all the unoccupied space in the packaging, including in the containment system.

<u>**Criticality Safety Index**: CSI = 0 ("N" : infinity).</u>

FIGURE 26.1

SCHEMATIC VIEW OF TRIGA ELEMENTS

TRIGA FUEL ELEMENT DIMENSIONS

–STANDARD
–*( ) THIN

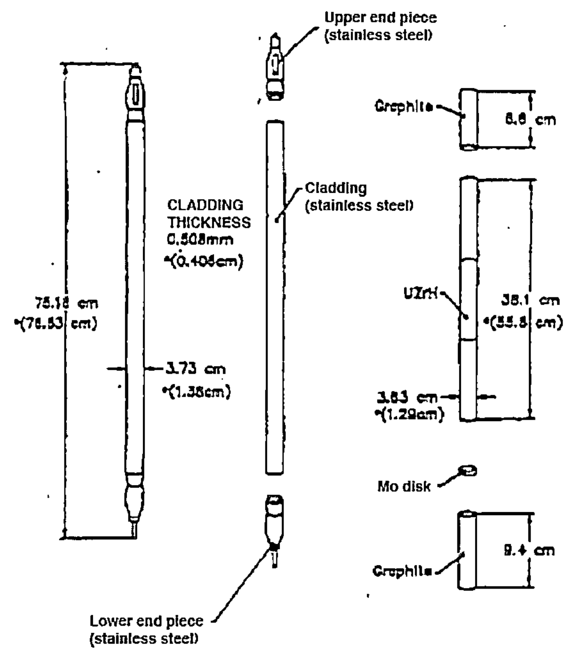
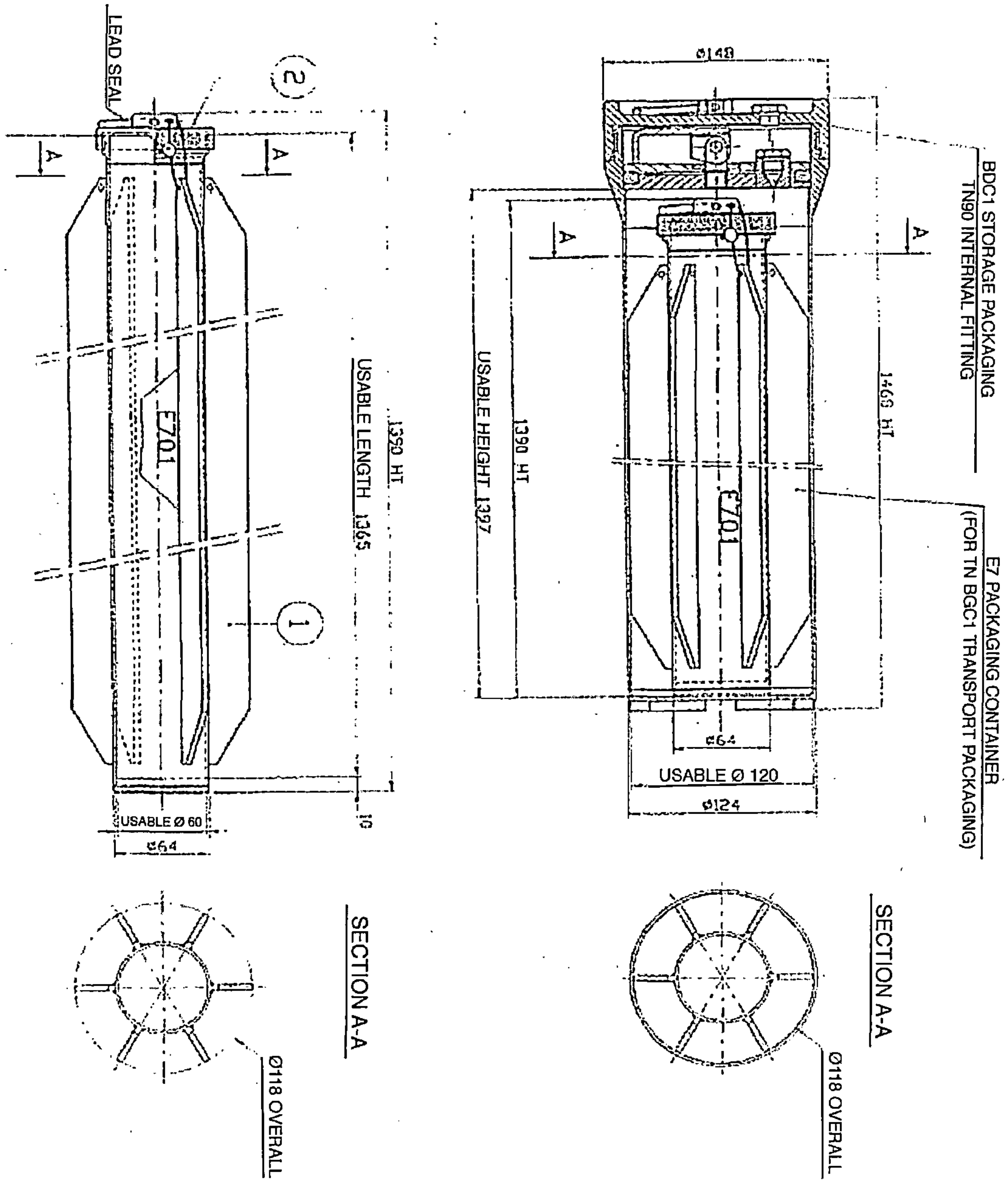


FIGURE 26.2

SCHEMATIC VIEW OF E7 TYPE CONTAINER



Note: dimensions are in millimetres.