Structural

1 **RAI 1-1**:

The kind of spruce wood to be used is not specified, but the properties of the spruce wood to be used are specified in detail as far as they are essential for the mechanical behavior of the shock absorbers.
Material data sheet WB-03-03 Rev. 4 and WPB-03-03 Rev. 2 (see enclosures 1 and 2) regulate the properties and the testing of the wood during manufacture. This includes sorting class (quality), density, moisture content and deformation energy.

2 **RAI 1-2**:

As already explained in an earlier answer report B-TA-3991-Rev.2 was prepared in an early design stage only to evaluate the angle for the 9 m slap down drop test with a cask model. At this time pine wood was intended to be used as shock absorber material instead of later spruce wood. As explained in an earlier answer this has no effect on the results of the calculations. It was also explained that for the slap-down drop only the properties of the wood in the side regions of the shock absorbers (balsa wood longitudinal) are of interest.
The values used in B-TA-3991 for this (9 – 15 MPa) were from BAM tests performed in the 1978 (we sent the curve already) and are in our opinion in good agreement with the new results of tests performed at BAM which are summarized in Table 4-30 of the SAR.

3 **RAI 1-3**:

Concerning the time yield limit values for lead given in Figure 59 of [Guruswamy 2000] we contacted Professor Guruswamy directly by e-mail and received the confirmation (see enclosure 3) that the stress axis has a typographical error and a factor of 100 is missing (it should be x98100). Therefore for the stress of chemical lead with 0.059% Cu at 10000 h a value of 40 x 98100 ≈ 4 MPa and for 100000 h a value of ≈ 3.4 MPa can be derived.
The values given in Table 4-9 of the SAR (2.0 MPa and 1.7 MPa) are therefore conservative values by a factor of 2.

The stresses in the packaging body under routine conditions of transport are calculated in Chapter 4.3.3 of the SAR and summarized in Table 4-26 for the outer shell. No stresses for the lead shielding were calculated because the lead is no structural material but encased by the outer and inner shell with no opportunity to move.
From the results of a vertical 9 m drop test with the 1:3 cask model at operation temperature (100 °C) where for an acceleration of approx. 100g no lead slump was observed after the test it can be concluded that for routine conditions of transport (2g) no lead deformation will occur.

4 **RAI 1-4**:

Please find enclosed the data sheet for the brazing material HTL 2 we intend to use (enclosure 4). This material is a high temperature nickel based brazing material. The melting range is **970 to 1000°C** which is much higher than the temperatures in the payload of the NCS 45. During normal conditions of transport the temperature is according to the SAR chapter 5 Table 5-11 max. **397°C**. For accident conditions the maximum temperature is **411°C** (Table 5-16), however, for containment analysis a 100% failure of the cans was assumed in chapter 6 Containment analysis.

The shear strength of the brazing material at room temperature is the same or even better than the base material, i. e., the properties of the brazing connection are comparable to a welded connection.

The margin of safety of the brazing against melting under HAC is **2.36**.

5 **RAI 1-5**:

Canning of high burn-up (> 45 GWd/MgU) fuel to maintain fuel configuration under HAC:

In the following the requirements towards the configuration of the fuel under HAC for criticality safety are discussed and it is shown that canning is not required to maintain the configuration and to guarantee criticality safety.

In calculation note RN-09-10-NCS-45 (see enclosure 5) it is shown that the dose rate limits for HAC are not reached by far even if the fuel is completely compacted. Hence, canning is not required to maintain the configuration and to guarantee that the dose rate limits are met under HAC.

Content 1.1:
For content 1.1 the inner components (baskets) according to drawings 0-090-108-00-00 and 0-090-112-00-00 are required (see SAR Table 2-2 and certificate Table 1.1). In the SAR sections 4.5.4.1 and 4.5.4.3 it is shown, that these inner components will remain intact during HAC. The safety is based on the limitation of the fissile material distribution per cm length of the zone containing fissile material.

For content 1.1 a fissile material distribution of not more than 16.4 g U-235 / cm length of the fissile material zone is specified (see SAR Table 2-2 and certificate Table 1.1).

Following assessment is based on the inner component according to drawing 0-090-112-00-00. This basket consists of 29 tubes with inner diameter 17 mm which accommodate the fuel.

In case of intact fuel the pellet diameter is conservatively 11 mm, giving a total fuel cross section of 27.56 cm^2. Taking into account the theoretical density of

10.96 g/cm^3 for the fuel and the certified enrichment of 5.3 wt. % the total fissile material per cm fissile length is 14.2 g U-235.
In case of a total failure of the fuel its scrap might fill the inner diameter of the tubes completely. The total cross section is hence 65.8 cm^2. We neglect the cladding completely. For the scrap conservatively a density of 4 g/cm^3 is assumed (UO_2 powder has a density of less than 3 g/cm^3). The total fissile material per cm fissile length is hence 12.3 g U-235.

In all cases the limit of 16.4 g U-235 / cm length of the fissile material zone is not reached by far. A failure of the fuel cladding does not invalidate the criticality analysis and the assumptions on which the specification of content 1.1 is based.

We would also like to remark that the criticality safety for content 1.1 does not take into account the steel components (tubes) of the baskets. These components reduce reactivity considerably. In the SAR section 8.6.1.5 it is shown, that even fuel rods with an enrichment of 7.0 wt. % U-235 in Uranium can be transported safely when the fuel is placed in steel tubes. Content 1.1 is defined to have an enrichment of 5.3 wt. %.

Content 1.2:
For content 1.2 the inner component (basket) according to drawings 0-090-111-00-00 **with a maximal inner diameter of 18 cm** is required (see SAR Table 2-3 and certificate Table 1.2). In the SAR section 4.5.4.2 it is shown, that this inner component will remain intact during HAC. The safety is based on the limitation of the fissile material diameter.

The criticality analysis does not take credit from the distribution of the fuel within the central tube of the basket. Optimal distribution and moderation with respect to maximal reactivity was assumed. A failure of the fuel during HAC does not invalidate the criticality analysis and the assumptions on which the specification of content 1.2 is based.

Content 1.3:
For content 1.3 several inner components (baskets) are specified. However, these baskets have no safety function but serve only as handling devices (see SAR chapter 8). The safety is based on the maximal mass of fissile mass in the package.

For the fuel itself a complete failure may occur, as the criticality analysis does not take credit from the distribution of the fuel within the cavity and/or any inner component. Optimal distribution and moderation with respect to maximal reactivity was assumed. A failure of the fuel during HAC does not invalidate the criticality analysis and the assumptions on which the specification of content 1.3 is based

Content 1.4:
For content 1.4 several inner components (baskets) are specified. However, these baskets have no safety function but serve only as handling devices (see SAR chapter 8). The safety is based on the limitation of the enrichment of the fuel in the package.

For the fuel itself a complete failure may occur, as the criticality analysis does not take credit from the distribution of the fuel within the cavity and/or any inner component. Optimal distribution and moderation with respect to maximal reactivity was assumed. A failure of the fuel during HAC does not invalidate the criticality analysis and the assumptions on which the specification of content 1.4 is based

Content 1.5:

For content 1.5 the inner component (basket) according to drawing 0-090-112-00-00 is required (see SAR Table 2-6 and certificate Table 1.5). In the SAR section 4.5.4.3 it is shown, that this basket will remain intact during HAC. The safety is based on the presence of the steel tubes of the basket.

For the fuel itself a complete failure may occur, as the criticality analysis does not take credit from the distribution of the fuel within the tubes (see SAR section 8.6.1.5). Optimal distribution and moderation with respect to maximal reactivity was assumed. A failure of the fuel during HAC does not invalidate the criticality analysis and the assumptions on which the specification of content 1.5 is based.

6 **RAI 1-6**:

The evaluation under item 2 applies for aluminum and stainless steel fuel cladding as well. Under HAC, the properties of the cladding are not relevant for criticality safety and for meeting the dose rate limits.

Shielding

7 **RAI 2-1:**

First, we would like to point out the Interim Staff Guidance – 6 (ISG 6) "Establishing minimum initial enrichment for the bounding design basis fuel assembly(s)". In this ISG it is stated that the neutron source is depending beside the burn-up also on the initial enrichment. ISG 6 states that the neutron source for fuel with a burn-up of 45 GWd/MgU and 3.3 wt. % enrichment is expected to be 70% higher than the neutron source for the same burn-up but 4.05 wt. % enrichment.

The intention of the shielding analysis was to provide bounding design basis source terms. Therefore, for the different burn-ups considered different enrichment values were used (see SAR Table 7-36). This enrichment values were chosen based on the guidance given in ISG 6. There, a value of 3.3% enrichment for fuel with burn-up of 45 GWd/MgU is specified as bounding design basis.

From the value given in ISG 6 a required enrichment per 10 GWd/MgU of 0.7 wt. % U-235 was derived. For the smallest burn-up of 10 GWd/MgU the enrichment was chosen as the fissile material limit of 1.0 wt. % U-235 and the bounding enrichments for the remaining burn-up values calculated from there (see Table 1).

For a burn-up of 80 GWd/MgU the derived bounding enrichment (1.0 + 7 x 0.7 = 5.9 wt. %) is greater than the maximum enrichment specified for contents 1.1 to 1.4, thus for all contents with a burn-up of greater 80 GWd/MgU the maximum enrichment of 5.3 wt. % is a conservatively bounding enrichment value.

The comparison of the enrichment values used in the SAR with the bounding enrichment values derived from ISG 6 are listed in Table 1.

Table 1: Derivation of the enrichment values used in the SAR based on ISG 6

Burn-up	Enrichment value in the SAR Table 7-36	Bounding enrichment value derived from ISG 6 with formula 1.0 + 0.07 * (burn-up – 10)
GWd/MgU	wt.% U-235	wt.% U-235
10	1.0	1.0
20	1.7	1.7
30	2.4	2.4
40	3.1	3.1
50	3.7	3.8
60	4.5	4.5
80	5.3[1)]	5.9
100	5.3[1)]	7.3
120	5.3[1)]	8.7

[1)] Upper limit of the licensed enrichment for contents 1.1 to 1.3

The very low enrichment values are hardly to be found in fuel of power reactors. Fuel with a low burn-up is expected to come from higher enriched fuel rods which have not been fully irradiated due to operational constraints. Hence, the enrichment values used in the SAR lead to an overestimation of the actual source terms.

For burn-up values of 80 GWd/MgU and above the enrichment is restricted to 5.3 wt. %. Based on ISG 6 it can be expected that the neutron source is much higher than if the derived bounding enrichment (e.g. 8.7 wt. % for a burn-up of 120 GWd/MgU) had been used, so that the SAR provides a highly conservative bounding case for the sources.

In the following it is shown, that the source terms given in the SAR Tables 7-37 and 7-41 follow mostly the well understood phenomenon that the gamma source term is linearly proportional to the burn-up and the neutron source is proportional to the fourth power of fuel burn-up. Then, for the values not complying with the phenomenon it is shown that the reasons for the (conservative!) deviations is the restricted enrichment.

Table 2 shows in the first two columns the ratios of the higher burn-ups in the SAR Table 7-37 to the next lower burn-up and in the following columns the ratios of the respective gamma source terms.

For cooling times of 5 years or more there is obviously a linear proportionality between burn-up and cooling time with a constant factor of 1. For shorter cooling times the constant factor is increasing to 1.15.

Table 2: Ratios of burn-ups and gamma source terms listed in the SAR Table 7-37

Considered burn-up values	Burn-up ratio	Gamma source strength ratio for cooling time (days)					
GWd/MgU / GWd/MgU	Cooling time days	120	180	365	730	1825	3650
20 / 10	**2,00E+00**	1,28E+00	1,36E+00	1,62E+00	1,79E+00	2,05E+00	2,06E+00
30 / 20	**1,50E+00**	1,13E+00	1,18E+00	1,29E+00	1,37E+00	1,51E+00	1,52E+00
40 / 30	**1,33E+00**	1,09E+00	1,12E+00	1,18E+00	1,24E+00	1,33E+00	1,35E+00
50 / 40	**1,25E+00**	1,07E+00	1,09E+00	1,14E+00	1,18E+00	1,25E+00	1,25E+00
60 / 50	**1,20E+00**	1,06E+00	1,07E+00	1,10E+00	1,13E+00	1,20E+00	1,20E+00
80 / 60	**1,33E+00**	1,10E+00	1,13E+00	1,19E+00	1,24E+00	1,32E+00	1,32E+00
100 / 80	**1,25E+00**	1,09E+00	1,11E+00	1,16E+00	1,19E+00	1,23E+00	1,22E+00
120 / 100	**1,20E+00**	1,07E+00	1,08E+00	1,10E+00	1,13E+00	1,16E+00	1,15E+00

Table 3 shows in the first two columns the ratios of the higher burn-ups in the SAR Table 7-41 to the next lower burn-up devided by a factor of 1.1 and in the following columns the 4th roots of the ratios of the respective neutron source terms. For burn-up values between 20 GWd/MgU and 80 GWd/MgU there is an excellent correlation.

Table 3: Ratios of burn-ups and 4th root of ratios of neutron source terms in Table 7-41

Considered burn-up values	Burn-up ratio / 1.1	4th root of neutron source strength ratio for cooling time (days)					
GWd/MgU / GWd/MgU	Cooling time days	120	180	365	730	1825	3650
20 / 10	**1,82E+00**	1,57E+00	1,58E+00	1,61E+00	1,64E+00	1,64E+00	1,63E+00
30 / 20	**1,36E+00**	1,28E+00	1,29E+00	1,30E+00	1,32E+00	1,32E+00	1,32E+00
40 / 30	**1,21E+00**	1,18E+00	1,19E+00	1,19E+00	1,20E+00	1,20E+00	1,20E+00
50 / 40	**1,14E+00**	1,14E+00	1,14E+00	1,15E+00	1,15E+00	1,15E+00	1,15E+00
60 / 50	**1,09E+00**	1,09E+00	1,09E+00	1,10E+00	1,10E+00	1,10E+00	1,10E+00
80 / 60	**1,21E+00**	1,21E+00	1,21E+00	1,22E+00	1,23E+00	1,22E+00	1,22E+00
100 / 80	**1,14E+00**	1,24E+00	1,25E+00	1,25E+00	1,25E+00	1,23E+00	1,22E+00
120 / 100	**1,09E+00**	1,26E+00	1,26E+00	1,25E+00	1,24E+00	1,21E+00	1,18E+00

The deviations from the expected proportionalities for neutron source terms for burn-up values above 80 GWd/MgU are due to the fact, that the enrichment is restricted to 5.3 wt. %. However, it has to be remarked that the calculated neutron source is higher than expected from the proportionality and hence provides a conservatively bounding case.

The source terms for low burn-ups have been calculated with conservatively low enrichment values. Enrichment values below 2.4 wt. % are hardly found in commercial nuclear fuel. However, for these values the relative burn-up for this fuel is still lower than the relative burn-up for higher burn-ups, thus leading to a lower neutron source than expected from the proportionality. This influence is shown in Table 3 in the line comparing 20 GWd/MgU / 10 GWd/MgU. Furthermore, this neutron source is of no real concern, because for low burn-up values the dose rate is governed by the gamma source term (see response to RAI 2-5 below).

8 **RAI 2-2:**

In calculation note RN-09-10-NCS-45 (see enclosure 5) axial peaking is analyzed.

9 **RAI 2-3**:

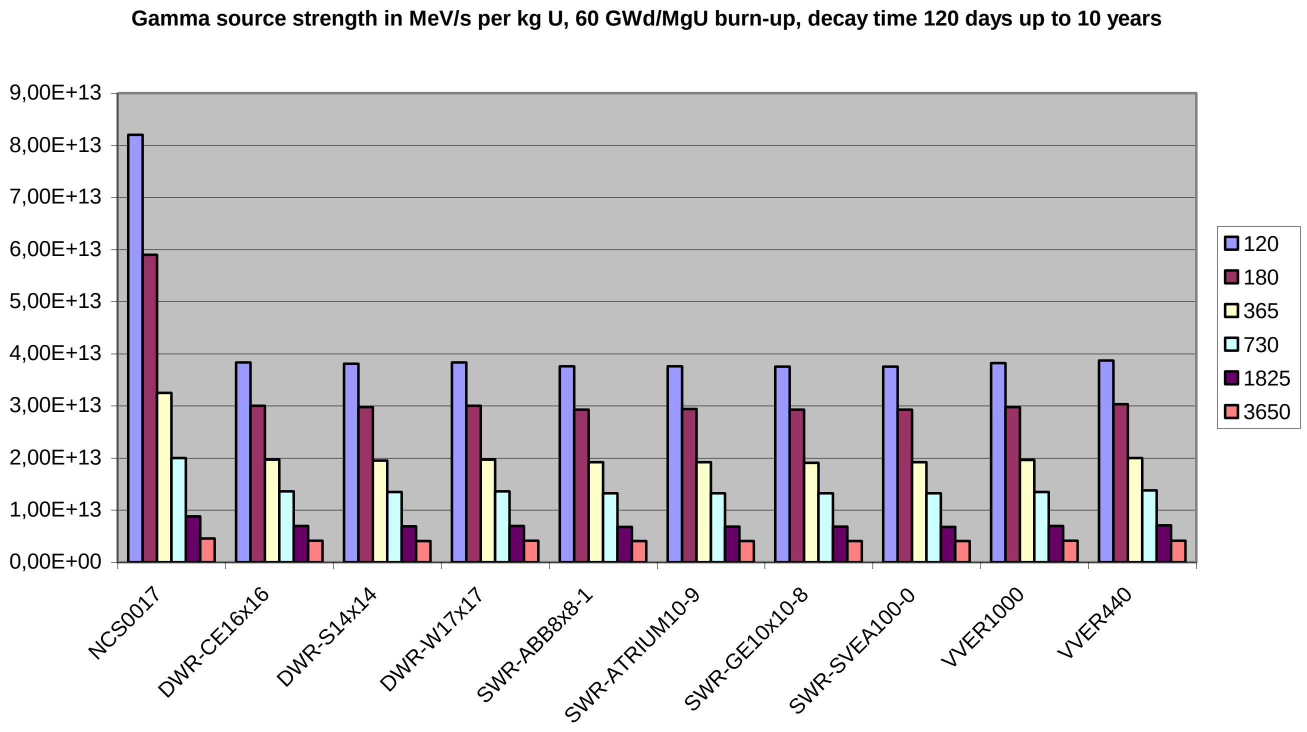
In calculation note RN-09-10-NCS-45 (see enclosure 5) the design basis contents specification is given based on permissible kgs of fuel as function of burn-up and cooling time. For this design basis contents specification source term calculation are carried out and the input data as well as the source spectra listed.

In detail following information is provided:

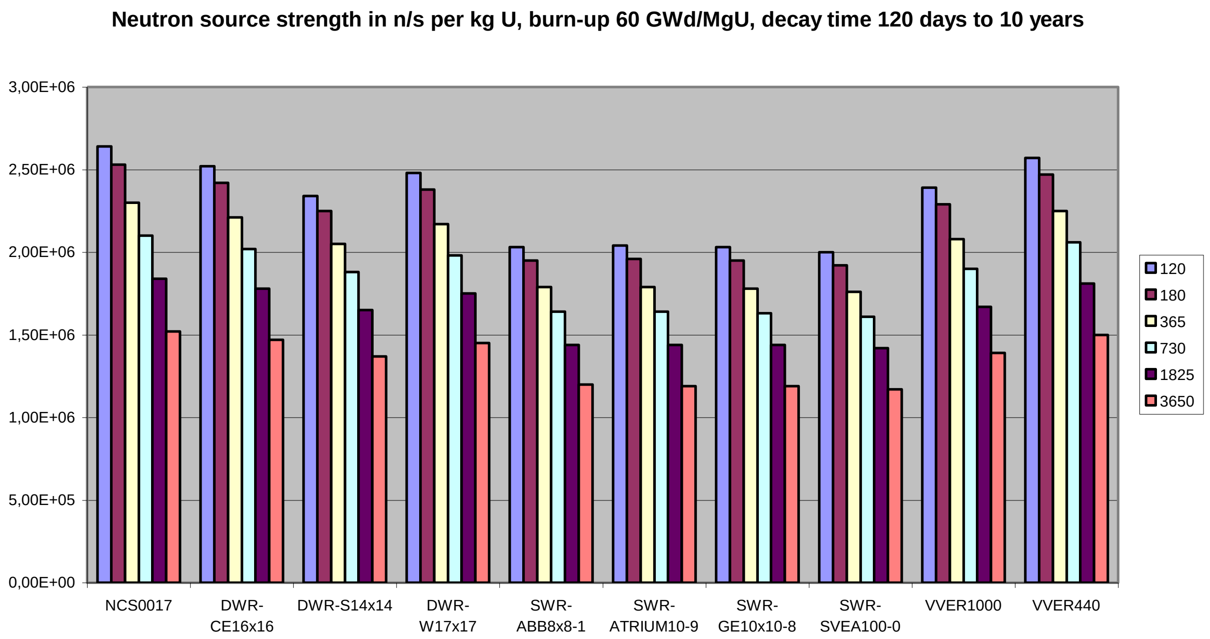
- Fuel assembly geometry
- Fuel power, number of cycles
- Burnable poison content data (in the moderator, see input file provided in RN-09-10-NCS-45)
- Fuel and moderator temperature (see input file provided in RN-09-10-NCS-45)
- Moderator density and soluble boron data (see input file provided in RN-09-10-NCS-45)
- Irradiation and depletion histories

The resulting source terms are compared with source terms calculated with the depletion code ORIGEN-ARP for burn-up values up to the currently implemented value of 70.5 GWd/MgU. For easy reading the respective figures from the SAR, Figure 7-15 and 7-16 are repeated. This figures show in the first multicolumn (designated with NCS 0017) the source terms used in the SAR and in calculation note RN-09-10-NCS-45 and in the following column the source terms for typical PWR, BWR and VVER fuel assemblies. This comparison shows that the source terms calculated here are conservative envelopes for the variety of existing fuel assembly designs.

SAR Figure 7-15 Comparison of gamma source strengths for PWR and BWR fuel assemblies with the source terms calculated in SAR Table 7-38 and RN-09-10-NCS-45



SAR Figure 7-16 Comparison of the total neutron source strengths for PWR and BWR fuel assemblies with the source terms calculated in SAR Table 7-41 and RN-09-10-NCS-45



From USA it is intended to transport irradiated fuel rods from the V. C. Summer and the Vogtle NPP. Enclosure 7a (report CE-09-573) contains the specification of the fuel rods to be shipped from V. C. Summer. The rods to be shipped from Vogtle NPP are covered by these data.

The covering data for the fuel rods are extracted from enclosure 7a:

Enrichment max.:	4.8435 wt. %
Burn-up max.:	72,054 MWd/MgU
In core time:	1599 days
Cooling time:	380 days (transport Nov. 01, 2010)
Fuel length:	365.8 cm
Total fuel weight:	15.82 kg

In enclosure 7b the gamma and neutron sources for the rods are given. They are summarized in Table 4.

Table 4: Gamma and neutron source terms for the fuel rods to be shipped from V. C. summer (November 2010), extract from enclosure 7b

Fuel rod	Calculated burn-up (MWd/MgU)	Gamma source (1/s)	Gamma source (MeV/s)	Neutron source (n/s)
D1	71017.8	1.51E+14	5.37E+13	6.19E+06
N1	71095.9	1.58E+14	5.58E+13	6.22E+06
A4	71105.1	1.50E+14	5.35E+13	6.22E+06
M4	74681.0	1.67E+14	5.95E+13	7.47E+06
Q4	71002.3	1.63E+14	5.71E+13	6.08E+06
B1	70765.7	1.50E+14	5.33E+13	6.12E+06
H4	72324.2	1.55E+14	5.53E+13	6.63E+06
D10	72325.1	1.54E+14	5.52E+13	6.63E+06
D8	72327.4	1.54E+14	5.51E+13	6.63E+06
B17	70828.5	1.51E+14	5.37E+13	6.13E+06
Sum (10 rods)		1.55E+15	5.52E+14	6.43E+07

For fuel rod M4 with the highest burn-up the source terms are calculated with the same input file as used for the tables in report RN-09-10-NCS-45. The results are given in Table 5.

In enclosure 7b 44 gamma energy groups were calculated, in report RN-09-10-NCS-45 only 18 energy groups are used. The comparison in Table 5 shows that the gamma source in 1/s calculated in enclosure 7b is approx. 5 % higher than calculated with the method used in report RN-09-10-NCS-45. However, the gamma source in MeV/s calculated in enclosure 7b is approx. 2 % lower than calculated with the method used in report RN-09-10-NCS-45. The neutron source calculated in enclosure 7b is also approx. 3 % higher than calculated with the method used in report RN-09-10-NCS-45.

Overall, the two methods can be considered to result in comparable source terms. The differences are well below the 10% uncertainties assumed in report RN-09-10-NCS-45.

Table 5: Comparison between the gamma and neutron source terms for the fuel rods to be shipped from V. C. Summer (November 2010) with the source terms evaluated in report RN-09-10-NCS-45

Fuel rod	Values from	Calculated burn-up (MWd/MgU)	Gamma source (1/s)	Gamma source (MeV/s)	Neutron source (n/s)
M4	V. C. Summer enclosure 7b	74681.0	1.67E+14	5.95E+13	7.47E+06
M4	Method used in RN-09-10-NCS-45	74681.0	1.59E+14	6.08E+13	7.22E+06

Table 6 shows the cycle burn-ups and the peaking factors. It can be noticed that the burn-up peaking factors decrease with increasing burn-up and that for the maximal burn-up peaking factors are 1.109. In RN-09-10-NCS-45 a peaking factor of 1.135 was assumed, covering the peaking factor to be assumed for the fuel rods to be shipped from V. C. Summer.

For the transport, the 10 V. C. Summer fuel rods would be classified in the design basis contents Table 7 below as permissible:

Burn-up: **less than 80 GWd/MgU**

Cooling time: **more than 365 days**

Permissible fuel mass: **less than 17.7 kg Uranium**

Table 6: Burn-up profiles for the fuel rods to be shipped from V. C. Summer

Row	Cold height, Mid-level (inch)	Cold Axial Row Thickness (inch)	Cycle 16 EOC (MWd/MgU)	Cycle 17 EOC (MWd/MgU)	Cycle 18 EOC (MWd/MgU)	Cycle 16 EOC rel. Burn-up	Cycle 17 EOC rel. Burn-up	Cycle 18 EOC rel. Burn-up
28	142.5	3	10387	20877	30192	0.391	0.411	0.437
27	139.5	3	12846	25320	35935	0.484	0.499	0.520
26	136.5	3	18502	36394	51632	0.697	0.717	0.747
25	134.5	1	19889	39081	55014	0.749	0.770	0.796
24	133.5	1	20263	40792	58162	0.763	0.804	0.841
23	132.5	1	20920	42142	59928	0.788	0.831	0.867
22	129	6	23238	46388	65317	0.875	0.914	0.945
21	123	6	25936	51049	71061	0.977	1.006	1.028
20	117	6	27296	53275	73647	1.028	1.050	1.065
19	111	6	27778	53552	73326	1.046	1.055	1.061
18	103.5	9	28145	54065	73794	1.060	1.066	1.068
17	94.5	9	28494	54511	74193	1.073	1.074	1.073
16	85.5	9	28819	54926	74585	1.085	1.083	1.079
15	76.5	9	29136	55338	74980	1.097	1.091	1.085
14	67.5	9	29453	55764	75391	1.109	1.099	1.091
13	58.5	9	29787	56218	75826	1.122	1.108	1.097
12	49.5	9	30119	56666	76268	1.134	1.117	1.103
11	40.5	9	30405	57039	76628	1.145	1.124	1.109
10	33	6	30464	57101	76642	1.147	1.125	1.109
9	27	6	30143	56613	76025	1.135	1.116	1.100
8	21	6	29090	54965	74019	1.095	1.083	1.071
7	15	6	26518	50751	68851	0.999	1.000	0.996
6	11.5	1	24034	46456	63550	0.905	0.916	0.919
5	10.5	1	23327	45096	61841	0.878	0.889	0.895
4	9.5	1	22692	42714	57675	0.854	0.842	0.834
3	7.5	3	20899	39502	53829	0.787	0.779	0.779
2	4.5	3	14110	26769	36604	0.531	0.528	0.530
1	1.5	3	10282	20086	28247	0.387	0.396	0.409

10 **RAI 2-4**:

We propose to exclude any irradiated solid nonfuel materials other than Zirconium alloys to be loaded together with the fuel from the DOT certificate. Any irradiated Zirconium alloys and any non-irradiated solid nonfuel material should be permitted alone or as load together with the fuel.

11 **RAI 2-5**:

In calculation note RN-09-10-NCS-45 (see enclosure 5) the design basis contents specification is given based on permissible kgs of fuel as function of burn-up and cooling time. Source spectra are given for all of the different design basis contents specifications. Shielding analyses are carried out for these design basis contents and documented in RN-09-10-NCS-45.

Therefore, we propose to describe the design basis in a conservative bounding way and to include the following Table 7 in the validation of the DOT certificate.

Table 7: Design basis content: permissible mass of fuel in kgs of heavy metal (identical to Table 1 of RN-09-10-NCS-45)

Burn-up GWd/MgU Cooling time days	Permissible mass of fuel in kgs of heavy metal					
	120	**180**	**365**	**730**	**1825**	**3650**
10	75.0	75.0	75.0	75.0	75.0	75.0
20	75.0	75.0	75.0	75.0	75.0	75.0
30	75.0	75.0	75.0	75.0	75.0	75.0
40	64.0	70.0	75.0	75.0	75.0	75.0
50	40.0	46.0	53.0	63.0	75.0	75.0
60	30.0	32.0	37.5	44.0	53.0	65.0
80	15.3	16.3	17.7	19.8	23.4	28.5
100	6.7	6.9	7.5	8.3	10.2	13.2
120	2.7	2.8	3.1	3.5	4.9	6.9

12 **RAI 2-6**:

A revised shielding analysis is provided in calculation note RN-09-10-NCS-45 (enclosure 5).

13 **RAI 2-7**:

The source terms as given in RN-09-10-NCS-45 (enclosure 5) were calculated with the code SAS2H. Input files used to calculate the ORIGEN/ARP libraries were adapted for the burn-up range up to 120 GWd/MgU.

14 **RAI 2-8**:

The thermal insulation layer (cement layer) was under accident conditions of transport completely replaced by vacuum, see RN-09-10-NCS-45 (enclosure 5) appendix 3. There is no neutron shielding provided by this layer during accident conditions of transport.

15 **RAI 2-9**:

For the criticality analysis different fuel geometries were used to reach conservative results:

- For an enrichment of 5.3 wt.% theoretical small fuel diameters were used because there is a tendency to higher reactivity for smaller fuel diameters. However, reactivity for homogeneous models is smaller than for heterogeneous models.
- For an enrichment of 3.4 wt.% theoretical small fuel diameters as well as fuel diameters up to 10 mm were analyzed (see the calculation note RN-09-03 Rev. 1 (enclosure 6)).
- For an enrichment of 7 wt.% theoretical large fuel diameters up to 17 mm were analyzed, as this proved to be the most reactive dimension.

With these analyses it is shown that arbitrary fuel diameters are criticality safe.

For the shielding analysis provided in RN-09-10-NCS-45 (enclosure 5) a standard pellet and fuel rod diameter was assumed.

16 **RAI 2-10**:

The units in Table 7-2 are the same as in Table 7-16 and correct. In Table 7-3 there is a writing error: the correct unit is µSv/h/(1/s) as in Table 7-17. We apologize for this writing error.

17 **RAI 2-11**:

We propose to limit the DOT certificate to full length fuel rods from commercial NPPs. A transport of sections of fuel rods is not foreseen for the next future from or to USA.

18 **RAI 2-12**:

Please see response to RAI 2-4.

19 **RAI 2-13**:

RAI 2-13 is identical to RAI 3-1. As this question relates to thermal properties, it will be answered under RAI 3-1.

20 **RAI 2-14**:

Lead slump was investigated in drop tests. In the SAR sections 4.5.3.5.8 and 4.5.3.7.8 the performance and result of drop tests with a specimen at maximum operating temperature is documented. SAR Fig. 4-16 and 4-17 show the longitudinal section of the specimen after all 19 drop tests including the drop test at MNOT were completed. Both figures show that there is no axial gap between the lead and the steel shell after the drop tests. Hence, lead slump can be excluded for the NCS 45.

Thermal

21 **RAI 3-1**:

In SAR Table 5-2 the maximum allowable decay heat is specified with 3000 W. Table 8 below lists the decay heat of the design base contents defined in Table 7. The Table shows that the decay heat of the design basis contents is in all cases below the maximum allowable decay heat.

Table 8: Thermal power for the design basis content listed in Table 7 in W

Burn-up GWd/MgU Cooling time days	Total thermal power in W					
	120	**180**	**365**	**730**	**1825**	**3650**
10	1215	818	368	165	41	23
20	1613	1163	608	293	83	50
30	1883	1410	788	398	128	75
40	1798	1505	938	495	173	113
50	1228	1095	758	491	225	143
60	987	826	596	396	196	156
80	571	489	343	228	124	103
100	279	237	174	120	74	66
120	123	107	82	61	46	46

Criticality

22 **RAI 4-1**:

The "Guide to Verification and Validation of the SCALE-4 Criticality Safety Software" can be used for SCALE5 and SCALE5.1 based on following reasons:

- The central KENO VI calculation module was not revised from SCALE4 to SCALE5
- The above mentioned guide provides a route for the V&V, but does actually not serve as only V&V document
- The V&V of the SCALE5 and SCALE5.1 criticality software is based on the installation verification, which has been carried out on the respective computers used for the safety analysis; and the V&V is especially based on recalculation of benchmarks, which is documented in the SAR NCS 0017 section 8.3.
- The criticality analyses have been checked and confirmed by the German competent authority by using an independently V&Ved SCALE5 criticality safety software system.

We forwarded the question whether the SCALE-4 guide could be used to Mr. Steve Bowman, Reactor Physics Group Leader of the Nuclear Science & Technology Division, ORNL, (e-mail: ScaleHelp@ornl.gov). His response was, "Yes, the report can be used with SCALE 5" (e-mail dated 22.09.2009, see enclosure 8).

23 **RAI 4-2:**

Enclosed please find the input files for the fissile material distributions 5.4156 g/cm, 26.2983 g/cm and 50.0981 g/cm for content 1.2 (enclosure 9, 10, and 11). The input files starting with "water_tube" contain the input for a theoretical diameter restriction to 18 cm with a tube consisting of "water". For the most reactive case the "water tube" is replaced by the "steel tube" required by the certificate of package approval in reality (enclosure 12). The comparison between the two calculations show, that the proof of content 1.2 is very conservative.
The enclosed Excel file "water fractions" (enclosure 13) shows the calculation of the respective water fractions. These values are based on the formulas given in SAR NCS 0017 section 8.5.3.1.
The naming convention for the files is
content no – restricting tube material – fuel – fuel rods/homogeneous – number of fuel rods – fuel mass per cm – water fraction – 1 – running number – fissile material distribution

24 **RAI 4-3**:

Enclosed please find the 3 input files (enclosure 14, 15, and 16) for the fissile material height of 52 cm. The enclosed Excel file "water fractions" (enclosure 13) shows the calculation of the respective water fractions.
The naming convention for the files is
content no – fuel – fuel rods/homogeneous – number of fuel rods – fuel mass per cm – water fraction – 1 – running number

25 **RAI 4-4**:

Content 1.5 was analyzed with model HET4; the title of RAI 5-7 Table 4 as well as Figure 7 and 8 of the response to RAI 5-7 should state HET4 and not HET1. We apologize for this error.

RAI 4-5:

Calculations in NCS 0017 for the array of packages are based on following assumptions:

- For contents 1.1, 1.2, 1.3 and 1.5 the cavity is dry with the exception that water to the extent of the H/U-235 ratio of 1 is present (see section 8.7.1).
- For content 1.3 the cavity is dry with the exception that water to the extent of the H/U-235 ratio of 100 is present (see section 8.7.2).
- Between the packages in variation calculations vacuum and water with different layer thicknesses are assumed (see Table 8-16). This Table shows that vacuum between the packages leads to maximal reactivity.
- The possible drying out of the thermal insulation layer is also accounted for by reducing the water content in the insulation material to zero (see Table 8-16).

Enclosed please find the input files for the most reactive case for ARRAY1 and ARRAY2 (see enclosure 17 and 18).

In additional calculations documented in calculation note RN-09-06-NCS-45 submitted to DOT/NRC with the first round RAIs the influence of water in the cavity was analyzed comprehensively. This analysis is based on highly theoretical assumptions concerning water ingress into the cavity requested by another competent authority during the validation process of the certificate and verifies the high safety margin of the NCS 45 package under accident conditions of transport.