SKI Report 02:52

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Radiation Shielding Assessment Using the SCALE Computer Code System

A demonstration based on an independent review of a real application

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This report concerns a study which has been conducted for the Swedish Nuclear Power Inspectorate (SKI) in co-operation with the Lithuanian Energy Institute (LEI). The conclusions and viewpoints presented in the report are those of the author and do not necessarily coincide with those of the SKI or the LEI.

SKI Project Number 98118

Abstract

The purpose of this project was to instruct a young scientist, Mr. Arturas Smaizys, from the Lithuanian Energy Institute (LEI) on how to carry out an independent review of a safety report. In particular, emphasis was to be put on how to use the personal computer version of the calculation system SCALE 4.3 in this process. Nuclear criticality safety together with radiation shielding from gamma and neutron sources were areas of interest. This report concentrates on shielding aspects while a separate report covers nuclear criticality safety.

The application was a proposed storage cask for irradiated fuel assemblies from the Ignalina RBMK reactors in Lithuania. The safety report contained various documents involving many design and safety considerations. A few other documents describing the Ignalina reactors and their operation were available. The time for the project was limited to approximately one month, starting "clean" with a SCALE 4.3 CD-ROM, a thick safety report and a fast personal computer.

The work of the author was originally planned for a total of 50 hours and was sponsored by the Swedish Nuclear Power Inspectorate (SKI). However, the task turned out to be more complicated and interesting which lead to much more involvement by the author. The results should be of general interest to Swedish authorities, in particular related to shielding where experience in using advanced computer codes like those available in SCALE is limited. The results of the project also indicate that there is a definite need for independent review of shielding assessments. This lesson was learned many years ago for nuclear criticality safety.

The project was carried out in co-operation with Mr. Smaizys from LEI. The participation of Mr. Smaizys was sponsored by the Swedish International Project (SIP) and by LEI. SIP supports improved nuclear fuel cycle safety in Eastern European countries. Some of the work was carried out in the office of SKI, who allocated a room and a fast computer to the project.

Several important results were obtained during the project. Concerning use of SCALE 4.3, it was confirmed that a young scientist, without extensive previous experience in the code system, can learn to use essentially all options. During the project, it was obvious that familiarity with personal computers, operating systems (including network system) and office software (word processing, spreadsheet and Internet browser software) saved a lot of time. Some of the Monte Carlo calculations took several hours. Use of three different computers (in a network) helped to get results within the time limit. Experience is valuable in quickly picking out input or source document errors. Understanding the basic theory and limitations behind the calculation methods require both studies and experience in using the methods. Experience in safety assessment is useful to sort out the important facts from all others and to identify important missing information. Bugs or undocumented limitations with potentially significant consequences must be expected in any large computer code system. It is believed that some bugs and undocumented limitations were found in SCALE 4.3.

The safety report appears to lead to correct conclusions. However, the importance of taking the axial burnup distribution into account when determining neutron source terms does not seem to be understood. The safety report seems to underestimate the dose variations on the surface of the cask. The dose reduction when moving away from the cask appears to be overestimated. Some of the information in the safety report is not clear.

Acknowledgements

It was a pleasure to work with Mr, Arturas Smaizys from the Lithuanian Energy Institute (LEI), In spite of his previous lack of experience with SCALE and shielding, he soon learned enough to initiate evaluations on his own and to participate in discussions. There is no doubt that, given time and support, Mr. Smaizys can become a prominent specialist on radiation shielding as well as on nuclear criticality safety.

Very much appreciated during the project was the support from Mr. Curt Bergman from the Swedish International Project (SIP), in particular related to radiation shielding and protection in general, as well as in obtaining information about Ignalina reactor design and operation.

The Swedish Nuclear Power Inspectorate (SKI), and Mr. Jan In de Betou in particular, supported the project in many ways and this was appreciated.

The availability of the computer code system SCALE is of enormous value to small countries like Sweden and Lithuania. Both the code developers at Oak Ridge National Laboratory (ORNL) and the sponsors at the United States Nuclear Regulatory Commission (NRC) are important in making SCALE a continuing success. During the project, ORNL answered questions by electronic mail about potential problems in SCALE and also proposed solutions. Documentation on verification and validation of SCALE had been sent by ORNL to the author during the last year and this turned out to be very useful during the project.

Contents

Radiation shielding assessment using the SCALE computer code system

1. Introduction

The report describes work related to a short project concerning use of SCALE 4.3 for assessment of radiation shielding of a cask containing irradiated nuclear fuel.

The purpose of the project was related to an independent review of a safety assessment reported to the relevant authority. A separate report covers nuclear criticality safety.

One of the conditions for the independent safety review was that calculations of neutron and gamma transport as well as of nuclear reactor irradiation and radioactive decay shall be carried out with the personal computer version 4.3 of the SCALE computer code package, developed at Oak Ridge National Laboratory (ORNL).

A real case is chosen for the independent review. The case involves storage and handling of irradiated RBMK fuel in casks in Lithuania.

The author has no previous experience with RBMK fuel or the cask used in the project. Further, the author has little experience in shielding and associated calculations.

2. Specifications related to shielding

2.1. Safety criteria

The only criteria used in this safety review is that the surface dose anywhere on the outside of the cask must not exceed 1000 μSv/hr.

The authority benefits from knowing that accidents and unexpected changes can be handled safely. Access to and experience with a modern computer code system like SCALE should increase the authority's confidence in being in control of the situation.

The results of an independent safety review can be used to grade priorities for inspection, quality assurance, emergency preparedness and response, etc. As an alternative to checking every detail and movement described in the application to the authority, the often few essential points can be identified and verified.

A safety review should not be restricted to the question of "safe enough". It may also be important in identifying the most efficient methods for control. Other hazards and interests such as environmental protection and preservation of natural resources may be related to shielding. Unnecessary control, resulting from conservative solutions, requires more resources and may also create worries that have no justification.

2.2. Radioactive material

The radioactive material cannot be specified exactly. Instead, it is a function of nuclear fuel specifications, nuclear reactor design and operation as well as time after irradiation in the reactor. Besides the fissionable material used to drive the reactor, other materials that also have been irradiated may be handled, shipped and stored together with the fuel.

The fuel is required to decay for at least five years before it is inserted into the cask.

The RMBK fuel consists of sintered, cylindrical pellets of UO_2 with a density between 10.4 and 10.5 g/cm^3. The pellet outside diameter is 1.152 cm and the height is 1.5 cm. In the radial centre of each pellet there is a hole with a nominal diameter of 0.2 cm. There is a tolerance for up to 0.23 cm diameter. At the ends of each pellet, there is a slight, spherical indentation. The safety report uses an average UO_2 density of 10.07 g/cm^3 to account for the holes and indentations.

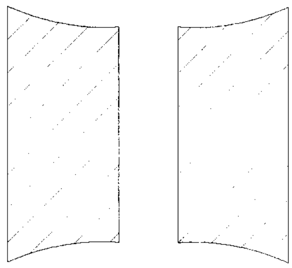


Fig 1. Fuel pellet cut

The uranium is enriched up to 2.4 weight-% ^{235}U. Preliminary interpretation of some documents leads to the conclusion that all fuel with this enrichment also has the burnable absorber erbium present. The average assay of erbium in the fuel pellet is given as between 0.41 and 0.43 wt-%.

Impurities and low concentrations of components of the compositions of fuel, cladding, spacers and other structural materials that are stored with the fuel can be important for shielding.

An example of impurities is ^{59}Co that is often present in stainless steel. The SCALE manual explains that small concentrations of this nuclide can lead to a significant contribution to the gamma dose.

Other nuclides that may be important are the uranium isotopes ^{234}U and ^{236}U. They depend on the enrichment. It is assumed here that no fuel is made from reprocessed uranium. If (when?) that becomes the case, there could be a significant change in the neutron source term.

2.3. Fuel rod and assembly

Each fuel rod contains a number of pellets of the type described above. A 1.36 cm outer diameter clad tube consisting of an alloy of zirconium (99 wt-%) and niobium (1 wt-%) surrounds the fuel pellets. The inside diameter of the clad can vary between 0.825 and 0.975 cm, leaving a gap between fuel and clad. The gap, holes and cavities in pellets all contain helium.

A fuel assembly consists of 18 fuel rods arranged in two concentrical circles around a central guide tube. This consists of a 1.5 cm outer diameter tube consisting of an alloy of zirconium (97.5 wt-%) and niobium (2.5 wt-%). The inner diameter is 1.25 cm.

In the inner circle, with a diameter of 3.2 cm, there are 6 equally spaced rods. In the outer circle, with a diameter of 6.2 cm, there are 12 equally spaced rods.

The fuel assembly is made up of two parts that are joined axially for a total length of almost 7 m active fuel. There is a short section in the middle of the assembly length where there is no fuel.

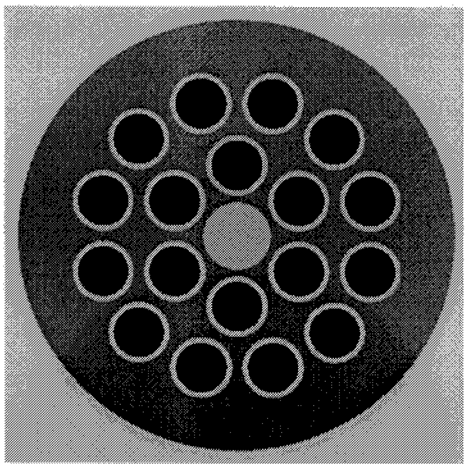


Fig 2. Fuel assembly in steel tube

Before storage in casks, the fuel assemblies are cut through this middle section, The bottom part is turned upside down before storage while the upper part is stored in the “normal” position. Another way of describing this is to require that the cut ends are always down in the cask. The active (fuel) length of each half assembly is 341 cm.

2.4. Reactor design and operation

The fuel has been irradiated in the Ignalina reactors. Important properties of this reactor type are that the neutrons are moderated with graphite and that the heat transfer medium (cooling) is water that is allowed to boil.

Some simplified facts about the reactor design and operation are required to make reasonable estimates of the compositions of irradiated components together with neutron and gamma source terms. The thermal heat generation of the irradiated fuel can also be estimated from the same information.

Since this was not a formal review and very little time was available for information search, some approximations were made about the reactor operation. Similarities between the RBMK fuel and BWR fuel in reactor operation were used to specify some of the water density and temperature conditions. Figures from a sample problem for SAS2H and BWR fuel in the SCALE manual were used to estimate water densities and temperatures of fuel and other components. These numbers were confirmed through comparisons with different information sources about operation of the Ignalina reactors.

The average water density for the RBMK was assumed to be 0.615 g/cm^3. This was based on information about the reactor operation and is higher than for BWR fuel. Low water density increases the radiation sources.

Here, the reactor design is simplified to a large number of high graphite blocks with a cylindrical vertical hole in the centre. Each block has square sides of 25 cm and the same height as the fuel assembly. In the cylindrical hole, the reactor fuel channel is positioned. It consists of a tube made of an alloy of zirconium (97.5 wt-%) and niobium (2.5 wt-%). Inside diameter is 8,0 cm and thickness is 0.4 cm. The small volume between the large graphite block and the tube is assumed to be completely filled with small graphite blocks. The fuel assembly fits tightly inside the fuel channel.

The figure shown above for the fuel assembly is for the storage in the cask, but a similar model describes the fuel assembly in the reactor.

The reactor operation is assumed to involve water entering the fuel channel from the bottom at a temperature of about 540 K and density of about 0.75 g/cm^3. The water is heated and allowed to boil. At the very top of the fuel, the water temperature is assumed to be about 554 K and the density about 0.48 g/cm^3. The average fuel temperature is assumed to be 840 K, the cladding temperature 620 K and the graphite 1023 K (maximum).

The maximum average burnup of the fuel covered by the safety report is 20 MWd/kgU. The boiling of water in the reactor leads to a reduction in water density higher up. This, together with neutron leakage, leads to a variable burnup rate along the length of the irradiated fuel assembly. A typical shape of this axial burnup is given in the safety report.

It is reasonable to assume that for a fixed burnup in MWd/kgU, lower enrichment will lead to higher source terms. This will be verified. Also, the presence of erbium may increase the sources.

It is assumed here that the fuel may be in the reactor in more than one cycle. The number of cycles as well as decay (cooling) times between cycles are assumed to vary.

2.5. The storage cask with internal basket

The geometry and materials in the cask have been simplified for the purpose of shielding design and review..

Radially there is an internal steel basket with an inner diameter of 147 cm. It is combined with the inner carbon steel cylinder of the cask wall to give a total thickness of 4.5 cm and an outside diameter of 156 cm. Further out, there is a concrete cylinder with thickness 35 cm. Finally, on the outside of the cask there is another 4 cm thick carbon steel cylinder.

Axially, the internal volume over the fuel region is 33 cm high. The bottom of the carbon steel basket is 5 cm thick. The bottom part of the cask consists of the same materials and dimensions as the radial wall. The lid is assumed to consist of a 33 cm thick layer of carbon steel.

The internal basket contains 102 stainless steel tubes, each with a fuel assembly inside. The steel tube has an outer diameter of 10.2 cm and a nominal wall thickness of 0.2 cm. The tubes are positioned in a triangular pitch of 12.5 cm.

The tube arrangement is such that the irradiated material will have a maximum outside radius of 62 cm.

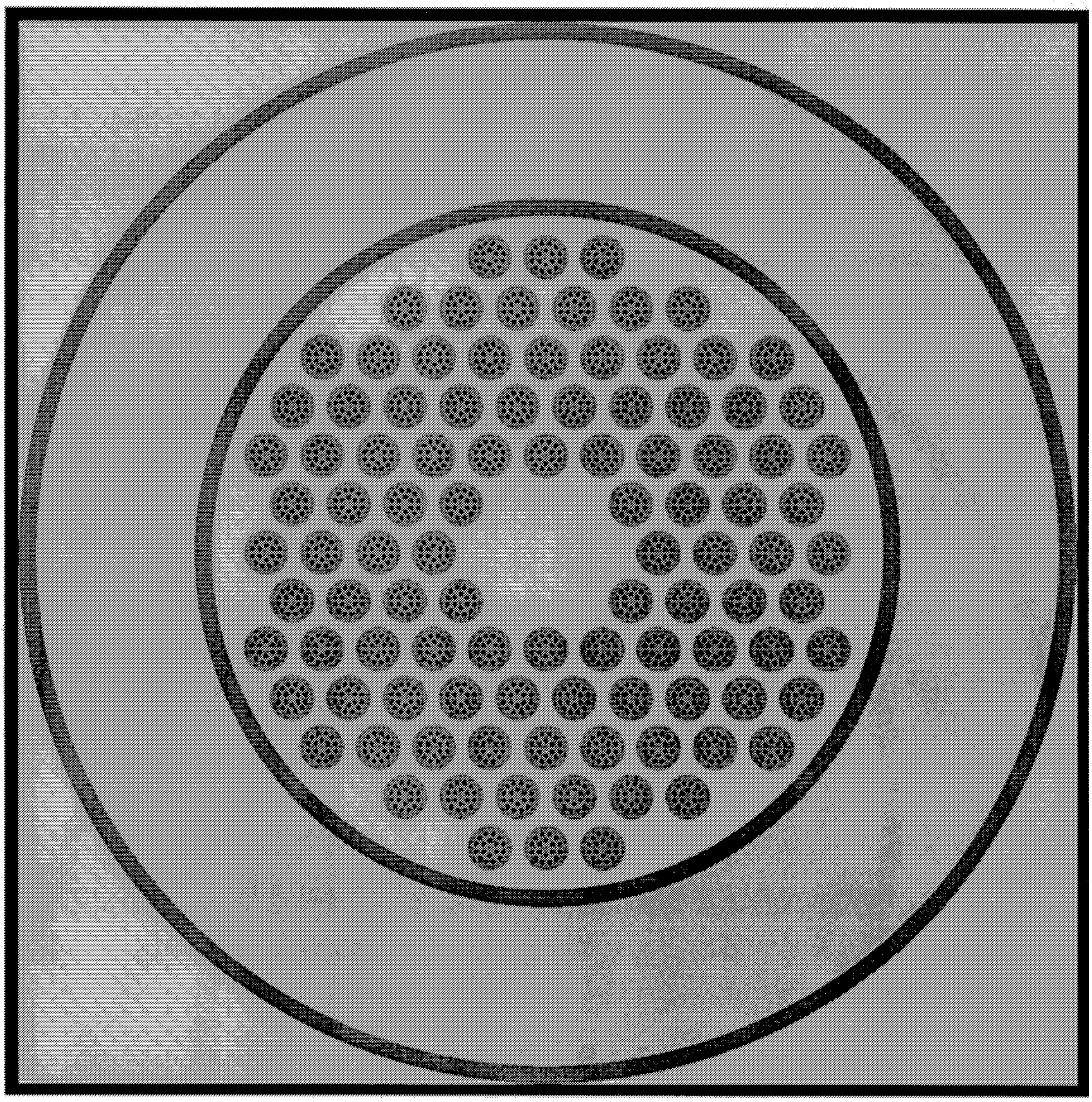


Fig. 3. Cask with half fuel assemblies in steel tubes - Plot from SCALE (KENO-Va)

3. Events related to shielding

3.1. Water inside cask

The safety report has assumed that there can be water inside the cask. This cannot happen during normal accident conditions of storage. The operations during loading and unloading of the cask may involve water inside the cask under normal or accident conditions.

The influence of the water will be to moderate, reflect and to absorb neutrons. The neutron multiplication will be higher with water than with dry fuel. Neutron multiplication and absorption are effects that work in opposite directions. If the neutron multiplication is not extremely high, the absorption effect will dominate.

The probability for water inside the cask has not been examined in this review.

With the high burnup assumed in the safety report, neutron multiplication will be low. Some simple calculations will be made with water to see the effect with homogenised fuel. A complete evaluation would require use of more detailed fuel geometry. Time did not allow such evaluation in this project..

3.2. Fuel and cask damage

The safety report does not discuss damage to the integrity of the fuel or to the fuel assembly. The basis for this assumption is not clear. However, it is likely that any severe damage to the fuel or cask will be noticed directly. Other shielding criteria should be used in such cases.

3.3. Operational mistakes

If the half assemblies are inserted into the cask with the wrong end first, the result could be increased doses. There is not enough information in the safety report to find out about this hypothetical event..

If the decay (cooling) time is shorter than five years, the doses could increase significantly.

If the average burnup is higher than 20 MWd/kgU the source terms will be higher. This leads to higher doses. This does not seem likely without increasing the enrichment.

If the enrichment ^{235}U is lower than 2 wt-%, the average burnup allowed must be reduced from 20 MWd/kgU. Otherwise the doses will become higher.

4. Calculation methods

4.1. SCALE 4.3 as a system

The SCALE 4.3 code package for personal computers (PC) is used. A CD-ROM version from early 1996 was used to install the package. Some upgrades, in particular the corrected version of the 44 group cross section library were downloaded from the WWW-site at Oak Ridge National Laboratory (ORNL).

No additional validation of the code package has been carried out. The RBMK fuel rods are similar to BWR fuel rods. During reactor operation, the assemblies are moderated by graphite. However, no graphite is present inside the fuel or in the cask.

4.2. SAS2H - Depletion, decay and radial dose calculations

Reactor burnup calculations were carried out to determine typical neutron and gamma source terms. The SCALE calculation sequence SAS2H was used for this purpose. SAS2H calculates the neutron spectrum for typical reactor conditions and then applies the depletion and decay code ORIGEN-S for generation of time-dependent cross sections.

Radial neutron and gamma doses are also obtained with SAS2H. The output from ORIGEN-S is used by the one-dimensional transport code XSDRNPM-S to calculate fluxes outside the cask. The XSDOSE code is used to convert the fluxes to doses at various distances from the outside surface.

4.3. SAS1 - Shielding analysis in simple geometry

The SAS1 calculation sequence contains some of the codes that are used in SAS2H. The source can be entered in the input or it can be copied from a previous SAS2H (or other ORIGEN-S) calculation. The XSDRNPM-S and XSDOSE codes are used for radial problems as in SAS2H. For axial problems, significant approximations are required.

4.4. QADS (SAS5) - Gamma shielding analysis with fast method

The QADS (SAS5) calculation sequence contains the code QADCGGP. This code is based on exponential reduction of fluxes together with build-up factors for secondary gamma effects. The source can be entered in the input or it can be copied from a previous SAS2H (or other ORIGEN-S) calculation. QADS is used for gamma doses primarily. There are options for calculating neutron doses as well.

An input limitation in QADS caused significant confusion. QADS uses the same material mixture input structure as other calculation sequences in SCALE. However, it turns out that only 20 materials (standard compositions) are allowed in a mixture. Since the radioactive contents of the cask were calculated with the SAS2H sequence, the input to the shielding calculation with SAS2H (the input to the last pass with XSDRNPM-S in SAS2H) was copied to SAS5. This input contained 22 materials, including actinides, two fission products and the stainless steel in the tubes containing the 102 half fuel assemblies.

The materials in this mixture self-shields the radiation from the neutron and gamma sources. Adding more materials, like the components of steel, reduces the external dose of the cask. However, it was found that inserting more than 20 materials in a mixture lead to unpredictable results. There was no error message or warning from the SCALE code system.

This limitation may be documented somewhere in the SCALE manual, but it has not been observed. The consequence was a serious under-prediction of the gamma radiation dose outside the cask. The dose was calculated a factor five low.

4.5. SAS4 - Shielding analysis in complicated cask geometry

The geometry of the cask containing RBMK fuel is quite complicated. The SCALE calculation sequence SAS4 uses the Monte Carlo code MORSE-SGC to calculate neutron and gamma transport in typical irradiated fuel cask geometry. The efficiency of the code is improved dramatically by the use of automatic biasing. This is based on adjoint calculations with the XSDRNPM-S code.

The MORSE-SGC Monte Carlo program is often quite complicated to use. In shielding calculation it is necessary to use biasing (weighting) to reduce the time the code spends tracking neutrons or gammas in less important regions. The SAS4 sequence uses a sophisticated technique for biasing. The adjoint flux is calculated with XSDRNPM-S using a dose detector as a source. This will only give 1-dimensional fluxes but that is often sufficient for casks with irradiated fuel.

A new feature in the SCALE 4.3 version of SAS4 allows axially distributed sources. This capability was taken advantage of, using the estimated sources from the given axial burnup distribution.

To find the dose distribution along the outside surface of the cask, the surface detector option cannot be used directly. This is because this detector type in SCALE 4.3 is automatically positioned around the symmetry line (cylinder axis) and plane (mid-plane of cask). For an axial problem, the surface detector must be centred around the centre of the lid or the bottom of the cask. For a radial problem, the surface detector must be centred at the mid-plane. The size of the detectors can be varied using the fractions FR1, FR2, FR3 and FR4.

Point detectors can be positioned in a more flexible way. However, they require much longer calculation times to get reliable statistics.

During calculations with point detectors, it was found that the doses near the symmetry line or plane were systematically underestimated. The reason is not known and the code developers at ORNL have been informed about a potential bias. However, ORNL has stated that the point detectors are not very efficient in general and recommends the use of surface detectors. This will become much easier with version 4.4 of SCALE since the surface detectors can be positioned in a more flexible way.

It turned out to be quite difficult to get good results for doses along the surface of the cask. Very long calculation times were required (ten million neutrons were often followed) but the results may still not always be very accurate.

4.6. SAS3 - General shielding analysis in complicated geometry

The sequence SAS3 using MORSE-SGC is available for general problems. This sequence was not used in this project.

5. Results

5.1. SAS2H - Compositions and sources - Average burnup distribution

The reactor operation needs to be significantly simplified to allow calculations of irradiated material compositions together with neutron and gamma source terms.

The first step is to simplify the fuel assembly. The 18 fuel rods are assumed to be part of a large array of rods in a triangular lattice. The cross section area of the assembly is assumed to have a diameter of 8.0 cm (including a little water outside the fuel). This cross section area is used to calculate the average triangular pitch (centre-to-centre separation) of fuel rods.

In the criticality safety assessment (using one of the CSAS sequences), the water area of the central tube is assumed to be divided between the 18 fuel rods. That is a sufficiently good approximation for nuclear cross section data preparation related to calculation of neutron multiplication factors. The triangular pitch for that purpose was calculated as 1.796 cm.

SAS2H allows a somewhat more sophisticated model. The neutron flux can be calculated in two steps. First, an infinite array model of fuel rods (similar to the criticality safety analysis) is applied. However, this time the central tube is assumed to take one position in the array (one tube and 18 rods). The fuel area is assumed to be a fraction 18/19 of the cross section area of the fuel assembly. This leads to a somewhat tighter pitch than in the criticality safety assessment. This pitch is 1.748 cm.

The result of this first step is a homogenised fuel region with nuclear cross sections that are reasonably adjusted for the neutron spectrum of the fuel rod array.

The second step is to model a reactor based on a large number of graphite blocks with fuel channels and homogenised fuel. The graphite block is transformed into a cylinder with the same cross section area.

The reactor model will then be based on a "cell" consisting of concentrical cylinders starting from the inside with the central tube, the homogenised fuel region, the reactor fuel channel and finally the graphite block. An infinite number of cells is created by a "white" boundary condition. The rest of the SAS2H calculations follows the standard procedure.

Some of the input parameters are geometry, fuel assembly power, time of reactor operation, time of decay (cooling), temperatures and densities. The results are shown in table 5.1.

Case Id	1.748 cm tri pitch. 102 half assemblies. 1 reactor cycle, 5 y cooling, H2O dens.=0.615 g/cm3. Temp: Graphite=1023K, Channel=620K, Clad=620K, H2O=547K, Fuel=840K	Neutron source 10^8 n/s	Gamma source 10^{16} ph/s
5.1.1	Ref. 20 MWd/kgU - Flat BU - 2.0 wt-% U-235	2.72	3.33
5.1.2	As 5.1.1 but 2.4 wt-% U-235	1.71	3.26
5.1.3	As 5.1.2 but with 0.41 wt-% erbium	1.87	3.25
5.1.4	As 5.1.1 but 50% of water density	2.99	3.34
5.1.5	As 5.1.1 but 22 MWd/kgU	4.41	3.76
5.1.6	As 5.1.1 but 18 MWd/kgU	1.61	2.92
5.1.7	As 5.1.1 but 30 MWd/kgU	20.91	5.62
5.1.8	As 5.1.1 but 10 MWd/kgU	0.14	1.47
5.1.9	As 5.1.1 but 30 MWd/kgU and water density 48 %	22.95	5.65
5.1.10	As 5.1.1 but 10 MWd/kgU and water density 75 %	0.13	1.47
5.1.11	As 5.1.1 but 10 reactor cycles (more updates of comp.)	2.52	3.30
5.1.12	As 5.1.1 but 5 cycles with 30 days cooling between	2.58	3.12
5.1.13	As 5.1.1 but 0.0059 kg Co-59 mixed with steel	2.71	3.58
5.1.14	As 5.1.1 but only half graphite density	4.62	3.43
5.1.15	As 5.1.1 but no graphite	6.06	3.53
5.1.16	As 5.1.1 but 1 cm H2O, density 0.76%, no graphite	2.27	3.33
5.1.17	As 5.1.1 but power increased from 1.503 to 1.737 MW	5.58	4.00

Table 5.1. SAS2H calculations of neutron and gamma sources

Cases 5.1.1 and 5.1.4 in table 5.1 show the sensitivity of the sources to a change in the water density during reactor operation. Another set of calculations, with identical materials and geometry and very similar temperatures, involved a wider range of temperatures. The exact results are not shown here, it is the normalised trends that are of interest, see figure 4.

It is important to study this type of parameters and to be able to explain them in terms of physics. Lower water density results in a harder neutron energy spectrum which in turn results in more high actinides. These are very strong neutron sources.

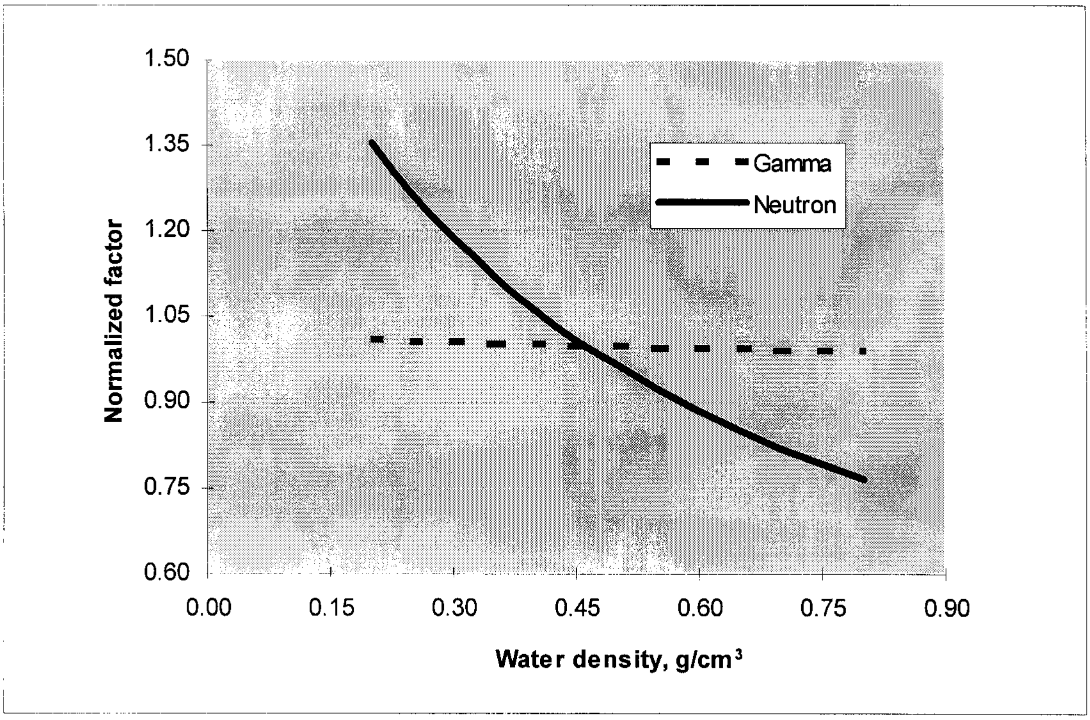


Fig 4. Neutron and gamma sources as functions of reactor operation water density

5.2. SAS2H - Compositions and sources - Realistic burnup distribution

The safety report includes a graph of the burnup distribution as a function of fuel assembly length. The graph was generated for fuel with an average burnup of about 14 MWd/kgU. To get an average burnup of 20 MWd/kgU, all numbers were increased by the same factor, about 1.4.

The burnup distribution means that the neutron and gamma sources will vary axially in the assembly. Since the fuel assemblies are cut into half before being inserted in the cask, the source distribution will depend on loading conditions. According to the safety report, all lower halves of fuel assemblies will be turned upside down.

The safety report assumes that the two halves of each original fuel assembly are loaded together in the same cask.

As was seen in the previous section, the average burnup is not a good parameter for determining sources, in particular neutrons. It seems as if the total sources will be higher with a realistic distribution than with an average (flat) burnup distribution.

Not only burnup, but also water density and temperatures of the different materials will vary axially (along the length of the fuel assembly).

To find out the influence of the burnup distribution on the total neutron and gamma sources as well as on the doses around the cask, the full assembly was divided into a number of zones. In the first effort (not reported here), different lengths of the zones were chosen. They were symmetrical only around the mid-plane of the full assembly. This caused problems later in the SAS4 model. Since the fuel assemblies are cut in half before storage and then each half has to be divided in two in the SAS4 model, the zones must be symmetrical when the assembly is divided into four parts.

Twelve equally long zones of the full assembly were chosen. For each zone, an average burnup was estimated from the given burnup distribution. The water density was assumed to vary linearly from 0.75 g/cm^3 at the inlet (bottom) of the assembly to 0.48 g/cm^3 at the outlet. This figure is based on reactor information about the maximum void fraction, 36%. In a BWR reactor the void fraction can be much higher.

It is important to note that the temperatures and other data about the reactor operation are based on preliminary data, some of them from the BWR reactor sample problem in the SAS2 section of the SCALE manual.

SAS2H calculations were made for the twelve different fuel assembly zones. The results are given in table 5.2.

Case Id	Axial zone - Distance from bottom [cm]		BU distr. MWd/ kgU	Av H2O density [g/cm3]	Total neutron source in zone n/s	Total gamma source in zone photons/s
	Low	High				
5.2.1	625.16	682.00	6.80	0.48	4.70E+05	8.08E+14
5.2.2	568.33	625.16	12.10	0.51	2.52E+06	1.53E+15
5.2.3	511.50	568.33	15.50	0.53	7.09E+06	2.04E+15
5.2.4	454.66	511.50	17.30	0.56	1.18E+07	2.33E+15
5.2.5	397.83	454.66	18.60	0.58	1.62E+07	2.54E+15
5.2.6	341.00	397.83	20.20	0.61	2.37E+07	2.81E+15
5.2.7	284.17	341.00	23.10	0.63	4.66E+07	3.33E+15
5.2.8	227.33	284.17	26.60	0.66	9.43E+07	4.01E+15
5.2.9	170.50	227.33	29.00	0.68	1.43E+08	4.46E+15
5.2.10	113.67	170.50	28.80	0.71	1.34E+08	4.42E+15
5.2.11	56.83	113.67	25.00	0.73	6.42E+07	3.67E+15
5.2.12	0.00	56.83	16.90	0.76	8.84E+06	2.25E+15
Total sources in cask					5.53E+08	3.42E+16

Table 5.2. SAS2H calculations of axial distribution of neutron and gamma sources

It is interesting to compare the distribution shapes of the burnup with those of the gamma and neutron sources. Normalised curves are shown in the figure on the next page. The water density distribution is not normalised (given in g/cm^3).

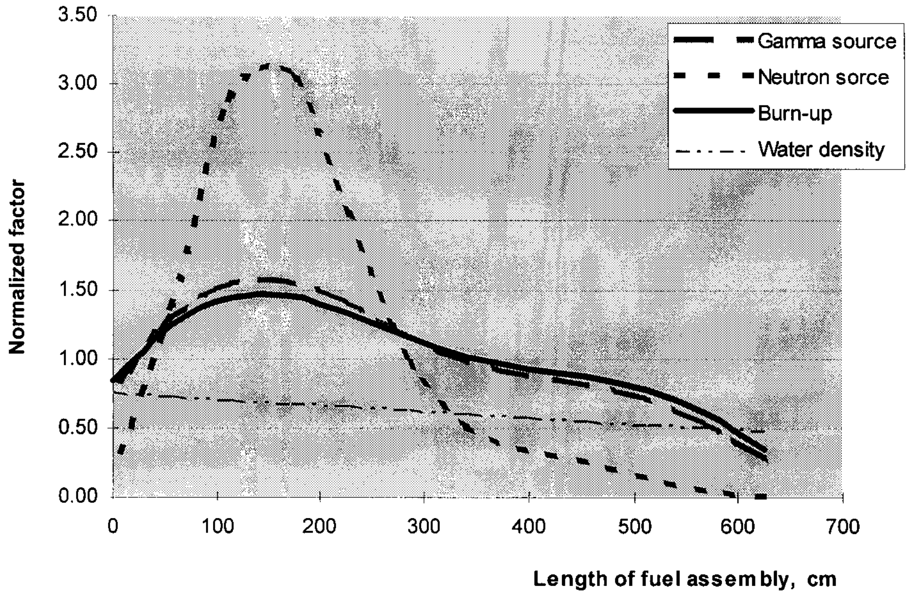


Fig 5. Normalised axial shapes of burnup, neutron source, gamma source and water density

5.3. SAS2H determination of radial neutron and gamma doses

If input for cask geometry is given, SAS2H automatically calculates radial gamma and neutron doses at various distances from the surface of the cask (0, 100, 200 and 400 cm). For a long cask, with all materials in cylindrical layers and with a homogeneous source, these doses should be quite accurate.

The dose results from some of the same calculations as were used to generate the sources given in table 5.1 are included in table 5.3. Some additional dose calculations have also been included.

Case Id	Description. 5 years decay	Source from table	Neutron source 10^8 n/s	Gamma source 10^{16} ph/s	Neutron doses mSv/h			Gamma doses mSv/h		
					0 m	1 m	2 m	0 m	1m	2m
5.3.1	2.0 % U-235	5.1.1	2.72	3.33	4.46	1.63	0.86	348	158	94.9
5.3.2	2.4 % U-235	5.1.2	1.71	3.26	2.87	1.05	0.55	326	149	89.1
5.3.3	Co-59 in SS	5.1.13	2.71	3.58	4.5	1.6	0.9	1025	465	278
5.3.4	22 MWd/kgU	5.1.17	5.58	4.00	9	3.3	1.8	438	199	119
5.3.5	Water in basket, not in cavity	5.1.17	5.58	4.00	6.8	2.5	1.3	409	186	112
5.3.6	Water in cavity	5.1.17	5.58	4.00	1	0.4	0.2	209	95	57.3
5.3.7	43.5 cm steel	5.1.17	5.58	4.00	258	91.6	47.6	2.4	1	0.6
5.3.8	33 cm steel	5.1.17	5.58	4.00	473	169	87.7	56.1	25.4	15.3
5.3.9	As 5.3.4 + 5 cm steel	5.1.17	5.58	4.00	4.5	1.6	0.9	68.3	31.3	18.8

Table 5.3. SAS2H calculations of neutron and gamma doses outside cask

5.4. SAS1 calculations of neutron and gamma doses

The SAS1 calculation sequence was used in a few test cases. For the radial gamma and neutron doses the results were identical to those that were obtained with SAS2H. This is not surprising since both sequences use the same data and computer codes.

The neutron and gamma energy group structures and source spectra from SAS2H calculations can be used directly by SAS1. Since the data from SAS2H (ORIGEN-S) are based on each fuel assembly, the total neutron and gamma sources for all 102 half assemblies must be given in the input to SAS1.

Axial calculations were also attempted. However, they required significant approximations (the cask has to be treated as a stack of infinite layers). A big effort would be required to get reasonable results and the idea was abandoned for this project.

The safety report uses this type of calculation to determine the axial (lid and bottom) neutron doses.

5.4. QADS (SAS5) calculation of gamma doses

As in SAS1, QADS can use the gamma energy group structure and source spectrum from SAS2H calculations directly. Since the data from SAS2H (ORIGEN-S) are based on a single fuel assembly, the total neutron and gamma sources for all 102 half assemblies must be given in the input to SAS5.

QADS is a very fast code. It was used to get a first impression and to compare with SAS4. QADCGGP was also used in the safety report for the cask.

Case Id	Source photons/s, Surface detectors on surface and point detectors near surface (1.5 cm). FR1=0.01 (radial) or 0.03 (axial)	Gamma doses μSv/h			
	Description.	Iron exp	Iron Dose	Concrete exp	Concrete Dose
5.4.1	Lid - Centre	60.9	61.4	Not applicable	Not applicable
5.4.2	Cask bottom - Centre	127	129	152	170
5.4.3	Radial mid-plane - Upper section	608	619	766	886

Table 5.4. QADS (SAS5) gamma doses - Homogeneous source distribution

Most of the calculations and time spent with QADS involved a code input limitation that was not checked by the code. The number of materials in a mixture is limited to 20.

In the comparison between SAS2H, SAS4 and SAS5, the number of materials in the homogenised contents of the cask was 22. The input was copied from a SAS2H output. The result was an under prediction of the dose by a factor higher than 5. Since this was a first attempt of using QADS, it took some time to rule out input errors to explain the low results. Removing a material or changing the order of input materials changed the results and this gave the clue to the mentioned code limitation.

5.6. SAS4 - Neutron and gamma doses from surface detectors

These calculations take a long time. Many calculations were carried out, many with less than sufficient statistical sampling. However, this seems to be the best tool in SCALE 4.3 for calculating gamma and neutron doses, in particular in the axial directions of the cask.

5.6.1. Axial variation of neutron and gamma sources - BUB and BUF arrays

To take the axial variation of the sources into account, the results given in table 5.2 were used.

BUB array - Distance [cm] from mid-plane		Sectors of assembly	BUF array - Fractions	
			Neutrons	Gammas
BUB(5) Top	170.5	Endpoint	0.00	0.00
BUB(4)	142.1	S1+S12	0.13	0.62
BUB(3)	85.2	S2+S11	0.92	1.06
BUB(2)	28.4	S3+S10	1.95	1.32
BUB(1) Centre	0.0	Midpoint	2.13	1.38
BUB(1) Centre	0.0	Midpoint	1.26	0.99
BUB(2)	28.4	S6+S7	1.38	1.05
BUB(3)	85.2	S5+S8	0.99	1.01
BUB(4)	142.1	S4+S9	0.63	0.95
BUB(5) Bottom	170.5	Endpoint	0.00	0.00

Table 5.5. SAS4 axial source profiles - Normal cask loading

BUB array - Distance [cm] from mid-plane		Sectors of assembly	BUF array - Fractions	
			Neutrons	Gammas
BUB(5) Top	170.5	Endpoint	0.00	0.00
BUB(4)	142.1	S12+S12	0.13	0.65
BUB(3)	85.2	S11+S11	0.93	1.06
BUB(2)	28.4	S10+S10	1.94	1.28
BUB(1) Centre	0.0	Midpoint	2.07	1.29
BUB(1) Centre	0.0	Midpoint	1.42	1.12
BUB(2)	28.4	S9+S9	1.51	1.13
BUB(3)	85.2	S8+S8	1.00	1.02
BUB(4)	142.1	S7+S7	0.49	0.85
BUB(5) Bottom	170.5	Endpoint	0.00	0.00

Table 5.6. SAS4 axial source profiles - Bottom parts of fuel assemblies

5.6.2. Neutron doses at upper radial surface - Average and mid-plane

Case Id	Source neutrons/s, Surface detectors FR1=1.0, FR2=FR3=0.7			Upper radial section - Neutron doses µSv/h, σ in %					
	Description.	Tables	10^8	0 m	σ	1 m	σ	2 m	σ
5.7.1	Flat - Proportional to BU	5.1.1	2.72	4.3	1	1.5	1	0.9	1
5.7.2	Flat - Based on realistic distribution	5.2.	5.53	8.6	1	3.1	1	1.7	1
5.7.3	Axial distribution. Whole assembly	5.2 and 5.5.	4.35	7.4	0	2.5	0	1.3	1
5.7.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	8.29	14.1	1	4.7	1	2.6	1
5.7.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	11.30	18.7	1	6.3	1	3.5	1

Table 5.7. Large surface detector neutron doses at "upper" radial section

Case Id	Source neutrons/s, Surface detectors FR1=0.01, FR2=FR3=0.007			Upper radial mid-plane Neutron doses µSv/h, σ in %					
	Description.	Tables	10^8	0 m	σ	1 m	σ	2 m	σ
5.8.1	Flat - Proportional to BU	5.1.1	2.72	4.5	3	1.7	3	0.96	3
5.8.2	Flat - Based on realistic distribution	5.2.	5.53	9.4	4	3.4	3	2.0	3
5.8.3	Axial distribution. Whole assembly	5.2 and 5.5.	4.35	14.0	3	3.6	2	1.7	3
5.8.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	8.29	26.5	3	7.1	3	3.4	3
5.8.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	11.30	28.9	3	8.6	3	4.1	3

Table 5.8. Small surface detector neutron doses at "upper" radial mid-plane

5.6.3. Gamma doses at upper radial surface - Average and mid-plane

Case Id	Source photons/s, Surface detectors FR1=1.0, FR2=FR3=0.7			Upper radial section - Gamma doses µSv/h, σ in %					
	Description.	Tables	10^{16}	0 m	σ	1 m	σ	2 m	σ
5.9.1	Flat - Proportional to BU	5.1.1	3.33	421	1	167	1	98.7	1
5.9.2	Flat - Based on realistic distribution	5.2.	3.42	433	1	172	1	101	1
5.9.3	Axial distribution. Whole assembly	5.2 and 5.5.	2.94	383	1	151	1	89.3	1
5.9.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	4.14	537	1	212	1	125	1
5.9.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	4.72	617	1	242	1	142	1

Table 5.9. Large surface detector gamma doses at "upper" radial section

Case Id	Source photons/s, Surface detectors FR1=0.01, FR2=FR3=0.007			Upper radial mid-plane Gamma doses µSv/h, σ in %					
	Description.	Tables	10^{16}	0 m	σ	1 m	σ	2 m	σ
5.10.1	Flat - Proportional to BU	5.1.1	3.33	455	4	203	3	121	3
5.10.2	Flat - Based on realistic distribution	5.2.	3.42	459	5	211	3	116	3
5.10.3	Axial distribution. Whole assembly	5.2 and 5.5.	2.94	569	4	226	3	120	3
5.10.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	4.14	787	4	394	24	161	3
5.10.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	4.72	826	10	318	3	175	3

Table 5.10. Small surface detector gamma doses at "upper" radial mid-plane

5.6.4. Neutron doses at lower radial surface - Average and mid-plane

Case Id	Source neutrons/s, Surface detectors FR1=1.0, FR2=FR3=0.7			Lower radial section - Neutron doses μSv/h, σ in %					
	Description.	Tables	10^8	0 m	σ	1 m	σ	2 m	σ
5.11.1	Flat - Proportional to BU	5.1.1	2.72	4.3	1	1.4	1	0.73	1
5.11.2	Flat - Based on realistic distribution	5.2.	5.53	8.6	1	2.7	1	1.5	1
5.11.3	Axial distribution. Whole assembly	5.2 and 5.5.	6.71	11.0	0	3.5	0	1.9	0
5.11.4	Axial distr. Lower part of assembly only. Normal position (cut end down)	5.2 and 5.6.	11.30	18.8	1	6.0	1	3.2	1
5.11.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	8.29	13.8	1	4.5	1	2.4	1

Table 5.11. Large surface detector neutron doses at "lower" radial section

Case Id	Source neutrons/s, Surface detectors FR1=0.01, FR2=FR3=0.007			Lower radial mid-plane Neutron doses μSv/h, σ in %					
	Description.	Tables	10^8	0 m	σ	1 m	σ	2 m	σ
5.12.1	Flat - Proportional to BU	5.1.1	2.72	5.0	5	1.6	3	0.89	3
5.12.2	Flat - Based on realistic distribution	5.2.	5.53	9.6	3	3.4	2	1.7	3
5.12.3	Axial distribution. Whole assembly	5.2 and 5.5.	6.71	15.8	3	4.7	2	2.4	3
5.12.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	11.30	28.6	3	8.5	2	3.9	3
5.12.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	8.29	25.9	3	6.9	2	3.1	3

Table 5.12. Small surface detector neutron doses at "lower" radial mid-plane

5.6.5. Gamma doses at lower radial surface - Average and mid-plane

Case Id	Source photons/s, Surface detectors FR1=1.0, FR2=FR3=0.7			Lower radial section - Gamma doses μSv/h, σ in %					
	Description.	Tables	10^{16}	0 m	σ	1 m	σ	2 m	σ
5.13.1	Flat - Proportional to BU	5.1.1	3.33	414	1	162	1	95.1	1
5.13.2	Flat - Based on realistic distribution	5.2.	3.42	436	1	170	1	100	1
5.13.3	Axial distribution. Whole assembly	5.2 and 5.5.	3.90	513	1	200	1	118	1
5.13.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	4.72	613	1	241	1	142	1
5.13.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	4.14	544	1	215	1	127	1

Table 5.13. Large surface detector gamma doses at "lower" radial section

Case Id	Source photons/s, Surface detectors FR1=0.01, FR2=FR3=0.007			Lower radial mid-plane Gamma doses μSv/h, σ in %					
	Description.	Tables	10^{16}	0 m	σ	1 m	σ	2 m	σ
5.14.1	Flat - Proportional to BU	5.1.1	3.33	461	4	202	3	114	3
5.14.2	Flat - Based on realistic distribution	5.2.	3.42	487	8	205	3	114	3
5.14.3	Axial distribution. Whole assembly	5.2 and 5.5.	3.90	579	4	254	3	131	3
5.14.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	4.72	775	4	322	3	180	3
5.14.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	4.14	755	3	291	3	170	3

Table 5.14. Small surface detector gamma doses at "lower" radial mid-plane

5.6.6. Neutron doses at lid - Average and centre

Case Id	Source neutrons/s, Surface detectors FR1=FR2=FR3=1.0			Upper axial section - Lid Neutron doses μSv/h, σ in %					
	Description.	Tables	10^8	0 m	σ	1 m	σ	2 m	σ
5.15.1	Flat - Proportional to BU	5.1.1	2.72	116	1	15.7	1	8.5	1
5.15.2	Flat - Based on realistic distribution	5.2.	5.53	236	1	32.0	1	11.1	1
5.15.3	Axial distribution. Whole assembly	5.2 and 5.5.	4.35	52.6	2	7.3	1	3.9	2
5.15.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	8.29	101	2	14.1	2	7.5	2
5.15.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	11.30	225	2	31.0	1	16.4	2

Table 5.15. Large surface detector neutron doses at lid

Case Id	Source neutrons/s, Surface detectors FR1=FR2=FR3=0.1			Upper axial - Centre of Lid Neutron doses μSv/h, σ in %					
	Description.	Tables	10^8	0 m	σ	1 m	σ	2 m	σ
5.16.1	Flat - Proportional to BU	5.1.1	2.72	160	3	35.6	1	12.5	2
5.16.2	Flat - Based on realistic distribution	5.2.	5.53	326	2	70.4	1	25.0	2
5.16.3	Axial distribution. Whole assembly	5.2 and 5.5.	4.35	66.6	6	17.0	2	5.6	4
5.16.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	8.29	129	5	30.2	2	11.2	4
5.16.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	11.30	304	4	71.5	2	25.5	3

Table 5.16. Small surface detector neutron doses at lid centre

5.6.7. Gamma doses at lid - Average and centre

Case Id	Source photons/s, Surface detectors FR1=FR2=FR3=1.0			Upper axial section - Lid doses μSv/h, σ in %					
	Description.	Tables	10^{16}	0 m	σ	1 m	σ	2 m	σ
5.17.1	Flat - Proportional to BU	5.1.1	3.33	52.9	1	8.9	1	6.4	1
5.17.2	Flat - Based on realistic distribution	5.2.	3.42	55.1	1	9.2	1	6.6	1
5.17.3	Axial distribution. Whole assembly	5.2 and 5.5.	2.94	7.8	3	1.4	3	1.0	3
5.17.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	4.14	11.1	3	2.0	3	1.4	3
5.17.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	4.72	17.0	3	3.0	3	2.1	3

Table 5.17. Large surface detector gamma doses at lid

Case Id	Source photons/s, Surface detectors FR1=FR2=FR3=0.1			Upper axial - Lid centre Gamma doses μSv/h, σ in %					
	Description.	Tables	10^{16}	0 m	σ	1 m	σ	2 m	σ
5.18.1	Flat - Proportional to BU	5.1.1	3.33	88.0	4	30.7	2	13.2	3
5.18.2	Flat - Based on realistic distribution	5.2.	3.42	84.3	4	31.7	2	13.3	3
5.18.3	Axial distribution. Whole assembly	5.2 and 5.5.	2.94	11.4	4	4.3	2	2.0	3
5.18.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	4.14	16.0	8	6.0	5	2.6	6
5.18.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	4.72	24.2	8	9.0	3	4.4	8

Table 5.18. Small surface detector gamma doses at lid centre

5.6.8. Neutron doses at bottom of cask - Average and centre

Case Id	Source neutrons/s, Surface detectors FR1=FR2=FR3=1.0			Lower axial section - Bottom Neutron doses μSv/h, σ in %					
	Description.	Tables	10^8	0 m	σ	1 m	σ	2 m	σ
5.19.1	Flat - Proportional to BU	5.1.1	2.72	2.1	2	0.31	8	0.16	2
5.19.2	Flat - Based on realistic distribution	5.2.	5.53	4.3	2	0.6	2	0.3	2
5.19.3	Axial distribution. Whole assembly	5.2 and 5.5.	6.71	2.2	2	0.4	11	0.2	5
5.19.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	11.30	3.2	3	0.46	2	0.26	3
5.19.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	8.29	1.3	4	0.21	10	0.1	4

Table 5.19. Large surface detector neutron doses at bottom of cask

Case Id	Source neutrons/s, Surface detectors FR1=FR2=FR3=0.1			Lower axial - Bottom centre Neutron doses μSv/h, σ in %					
	Description.	Tables	10^8	0 m	σ	1 m	σ	2 m	σ
5.20.1	Flat - Proportional to BU	5.1.1	2.72	3.1	5	0.66	3	0.22	5
5.20.2	Flat - Based on realistic distribution	5.2.	5.53	3.3	10	0.71	4	0.24	7
5.20.3	Axial distribution. Whole assembly	5.2 and 5.5.	6.71	6.5	4	1.4	3	0.49	4
5.20.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	11.30	4.3	6	1.1	5	0.38	7
5.20.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	8.29	1.7	11	0.41	14	0.14	10

Table 5.20. Small surface detector neutron doses at cask bottom centre

5.6.9. Gamma doses at cask bottom - Average and centre

Case Id	Source photons/s, Surface detectors FR1=FR2=FR3=1.0			Lower axial section - Bottom Gamma doses μSv/h, σ in %					
	Description.	Tables	10^{16}	0 m	σ	1 m	σ	2 m	σ
5.21.1	Flat - Proportional to BU	5.1.1	3.33	76.2	1	13.2	1	9.5	2
5.21.2	Flat - Based on realistic distribution	5.2.	3.42	78.4	1	13.6	1	9.7	1
5.21.3	Axial distribution. Whole assembly	5.2 and 5.5.	3.90	22.9	2	4.8	2	3.4	3
5.21.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	4.72	25.4	2	5.3	2	3.7	2
5.21.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	4.14	17.8	3	3.8	3	2.7	3

Table 5.21. Large surface detector gamma doses at cask bottom

Case Id	Source photons/s, Surface detectors FR1=FR2=FR3 =0.1			Lower axial - Bottom centre Gamma doses μSv/h, σ in %					
	Description.	Tables	10^{16}	0 m	σ	1 m	σ	2 m	σ
5.22.1	Flat - Proportional to BU	5.1.1	3.33	111	3	43.2	2	18.8	3
5.22.2	Flat - Based on realistic distribution	5.2.	3.42	111	3	44.8	2	18.8	3
5.22.3	Axial distribution. Whole assembly	5.2 and 5.5.	3.90	32.4	3	13.8	1	6.3	2
5.22.4	Axial distr. Lower part of assembly only. Normal position	5.2 and 5.6.	4.72	36.6	7	14.6	4	7.1	9
5.22.5	Axial distr. Lower part of assembly only. Inserted the wrong way	5.2 and 5.6.	4.14	27.6	11	9.9	4	4.6	7

Table 5.22. Small surface detector gamma doses at cask bottom centre

5.7. SAS4 - Neutron and gamma doses from point detectors

A limited selection of results obtained using point detectors are given in this section. All cases involve the normal situation with a complete RBMK fuel assembly cut into two parts. The top and bottom ends of each assembly face upwards while the cut surfaces are on the bottom of the cask. A realistic source distribution is used. The standard deviations are very large in some cases. Recalculation of a few cases using much better statistics resulted in very different results, though within two standard deviations.

5.7.1. Gamma doses from point and surface detectors

Case Id	Source photons/s, Surface detectors on surface and point detectors near surface (1.5 cm). FR1=0.01 (radial) or 0.03 (axial)			Gamma doses μSv/h, σ in %			
	Description.	Tables	10^{16}	Surface	σ	Point	σ
5.23.1	Lid - Centre	5.2 and 5.6.	2.94	11.4	4	10.3	10
5.23.2	Cask bottom - Centre	5.2 and 5.6.	3.90	32.4	3	33.3	17
5.23.3	Radial mid-plane - Upper section	5.2 and 5.6.	2.94	569	4	479	4
5.23.4	Radial mid-plane - Lower section	5.2 and 5.6.	3.90	579	4	525	5

Table 5.23. Small surface and point detector gamma doses - Distributed source

5.7.2. Radial gamma and neutron doses from point detectors near cask surface

The point detectors could not be positioned on the radial surface of the cask. For some numerical reason, the MORSE Monte Carlo program generated errors while tracking gammas or neutrons. When the detectors were positioned 2.0 cm away from the surface, there were no problems. This extra distance reduces the doses very little

Case Id	Detector position	Gamma doses μSv/h, σ in % distributed source			Neutron doses μSv/h, σ in % distributed source		
	cm	10^{16} Photons/s	Dose	σ	10^{8} Neutrons/s	Dose	σ
5.24.1	230	2.94	0.08	18	4.35	1.5	16
5.24.2	220	2.94	0.2	13	4.35	2.2	14
5.24.3	210	2.94	2.3	15	4.35	1.9	13
5.24.4	200	2.94	12.3	11	4.35	0.71	8
5.24.5	180	2.94	67	18	4.35	0.89	6
5.24.6	150	2.94	154	8	4.35	1.5	4
5.24.7	100	2.94	323	5	4.35	5.4	2
5.24.8	75	2.94	390	4	4.35	7.4	2
5.24.9	50	2.94	449	4	4.35	10	2
5.24.10	20	2.94	445	4	4.35	12	2
5.24.11	0	2.94	487	4	4.35	13.1	2
5.24.12	0	3.90	520	6	6.71	14	2
5.24.13	-20	3.90	494	2	6.71	14.3	2
5.24.14	-50	3.90	491	2	6.71	13	2
5.24.15	-75	3.90	490	2	6.71	11.8	2
5.24.16	-100	3.90	484	3	6.71	10	2
5.24.17	-150	3.90	287	8	6.71	5.3	10
5.24.18	-180	3.90	120	4	6.71	2.9	10
5.24.19	-200	3.90	25.2	5	6.71	0.93	6
5.24.20	-210	3.90	2.8	10	6.71	0.16	12

Table 5.24. Point detector radial gamma and neutron doses - Distributed sources

5.7.3. Axial gamma and neutron doses from point detectors on cask lid

The point detectors could be positioned directly on the surface of the cask. The same distributed gamma and neutron sources as were used in the calculations of the upper radial surface doses were used. The detector position is given as the radius from the axis of the cask.

Case Id	Detector position	Gamma doses μSv/h, σ in % distributed source			Neutron doses μSv/h, σ in % distributed source		
	cm	10^{16} Photons/s	Dose	σ	10^{8} Neutrons/s	Dose	σ
5.25.1	0	2.94	10	20	4.35	70.7	15
5.25.2	20	2.94	9.1	8	4.35	66.9	3
5.25.3	40	2.94	8.5	9	4.35	60.2	2
5.25.4	60	2.94	6	8	4.35	49.1	2
5.25.5	70	2.94	5.3	8	4.35	38.8	2
5.25.6	80	2.94	2.8	6	4.35	25.7	2
5.25.7	100	2.94	0.1	13	4.35	6.6	3
5.25.8	110	2.94	0.02	10	4.35	2.3	4

Table 5.25. Point detector neutron and gamma doses on cask lid - Distributed source

5.7.4. Axial gamma and neutron doses from point detectors on cask bottom

The point detectors could be positioned directly on the surface of the cask. The same distributed gamma and neutron sources as were used in the calculations of the lower radial surface doses were used. The detector position is given as the radius from the axis of the cask.

Case Id	Detector position	Gamma doses μSv/h, σ in % distributed source			Neutron doses μSv/h, σ in % distributed source		
	cm	10^{16} Photons/s	Dose	σ	10^{8} Neutrons/s	Dose	σ
5.26.1	0	3.9	32.6	19	6.71	3.5	10
5.26.2	20	3.9	30.3	5	6.71	3	3
5.26.3	40	3.9	30.1	23	6.71	2.4	2
5.26.4	60	3.9	15.8	4	6.71	1.9	5
5.26.5	70	3.9	17.9	12	6.71	1.5	3
5.26.6	80	3.9	15.3	6	6.71	0.9	3
5.26.7	100	3.9	3.3	5	6.71	0.22	5
5.26.8	110	3.9	0.98	6	6.71	0.08	11

Table 5.26 Point detector neutron and gamma doses on cask bottom. Distributed source

5.8. SAS4 - Calculations using a detailed model of the contents

In the previous calculations, the fuel and steel tubes were homogenised into a single body. The radiation self-shielding of this material is very strong. In the axial direction, the steel tubes may not be as effective with a realistic model as with the homogenised model. To get a better understanding of such effects, further calculations are required. Time did not allow such calculations during this project.

6. Discussion of safety report and results

6.1. Summary of safety report

The radiation shielding assessment in the safety report is based on calculations using ORIGEN for source terms, QADCGGP for gamma doses, ANISN for neutron doses and to a limited extent DORT for finding the neutron dose distribution along the surface of the cask.

The safety report acknowledges that there are uncertainties in the information about the reactor operation. This leads to uncertainties in the source terms. A correction factor of 1.5 is used to multiply the calculated neutron source term from ORIGEN.

The safety report also contains source terms from another document that was not available during this project. The source data were given as a function of energy groups. However, the group boundaries were not clearly specified and this information could not be used.

The basic conclusion of the safety report is that the radiation doses on the surface of the cask are less than 1000 μSv/h.

6.2. Comparison of results from safety report and from this project

Some of the results given by the safety report are summarised in tables 6.1 (source terms), 6.2 (gamma doses) and 6.3 (neutron doses).

Results from section 5 of this report are included as well, for comparison.

The somewhat cryptic descriptions in the graphs and tables should be interpreted as follows.

1. "BU" is short for BurnUp
2. "Di BU" is short for **Di**stributed "**B**"urn**U**p and indicates that the known burnup (BU) distribution was used to calculate the gamma and neutron sources.
3. "Di src" is short for **Di**stributed **s**ou**rc**e and indicates that the doses were calculated using a source distribution determined from the burnup distribution. It is important to understand that, in particular for neutrons, the source distribution is not proportional to the burnup distribution.
4. "Flat BU" is short for **Flat** **B**urn**U**p and indicates that the average burnup was used to calculate the gamma and neutron sources.
5. "Flat src" is short for **Flat** **s**ou**rc**e and indicates that, even when the distributed burnup is used to calculate the sources, a flat source (average source, not average burnup) is used in the calculations.

Case Id	2.0 wt-% U-235. 20 MWd/kgU average burnup.	Neutron source 10^8 n/s	Gamma source 10^{16} ph/s
6.1.1	ORIGEN calculations. Neutron source multiplied by 1.5	3.56	2.75
6.1.2	Reference to safety report - details unknown	1.62	1.98
6.1.3	SAS2H - 20 MWd/kgU. Flat burnup (BU)	2.72	3.33
6.1.4	SAS2H - 20 MWd/kgU. Realistic burnup	5.53	3.42

Table 6.1. Neutron and gamma sources from safety report and from this project

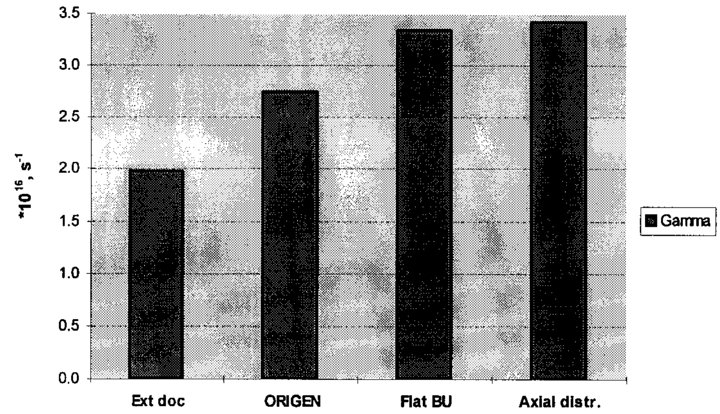


Fig. 6. Comparison of gamma source calculations

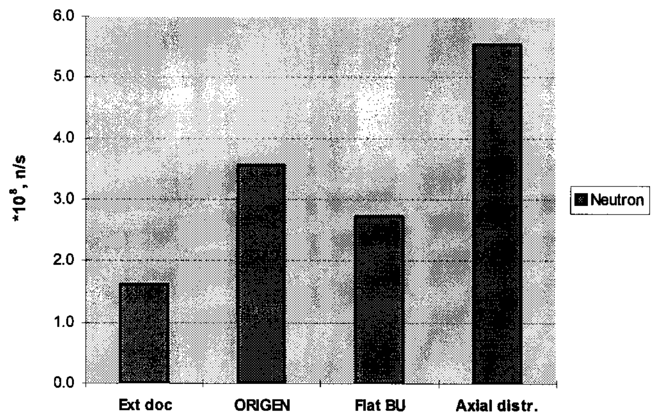


Fig. 7. Comparison of neutron source calculations

Case Id	Source origin	Doses μSv/h								
		Cask lid			Cask side wall			Cask bottom		
		0 m	1 m	2 m	0 m	1 m	2 m	0 m	1 m	2 m
6.2.1	QAD - Flat ORIGEN source	71	41	23	566	302	197	130	80	41
6.2.2	QAD - External reference. Flat	20	12	8	384	199	127	90	54	29
6.2.3	SAS4 - Hom. BU. Flat source	88	31	13	461	202	114	111	43	19
6.2.4	SAS4 - Real. BU. Flat source	84	32	13	487	205	114	111	45	19
6.2.5	SAS4 - Real. BU. Axial source	11	4	2	579	254	131	32	14	6
6.2.6	SAS5 - Real. BU. Flat source	61	-	-	608	-	-	127	-	-
6.2.7	SAS2H - Hom. BU. Flat source	-	-	-	348	158	95	-	-	-

Table 6.2. Gamma doses from safety report - Homogeneous source distribution

Fig. 8. Gamma doses at cask wall (radial)

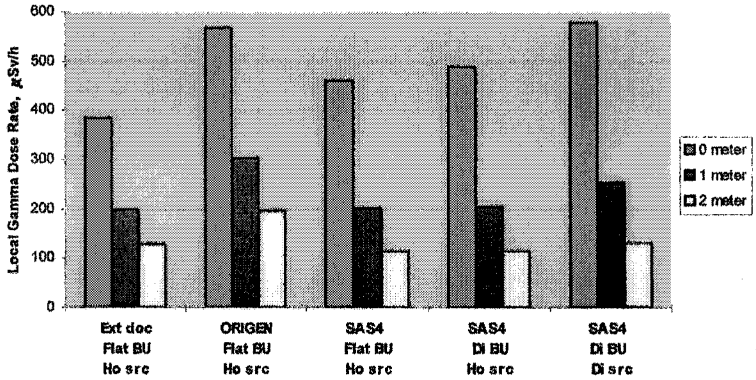


Fig. 9. Gamma doses at cask lid (axial)

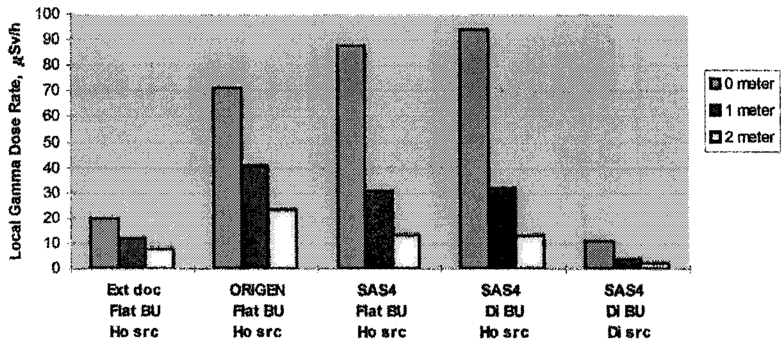
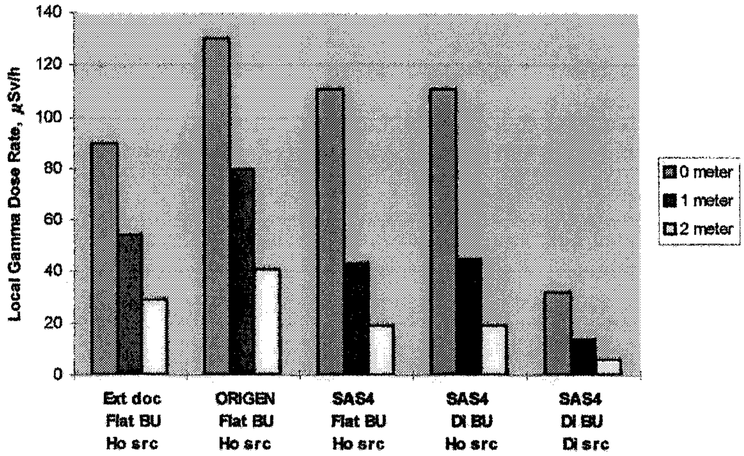


Fig. 10. Gamma doses at cask bottom (axial)



Case Id	Source origin	Doses µSv/h								
		Cask lid			Cask side wall			Cask bottom		
		0 m	1 m	2 m	0 m	1 m	2 m	0 m	1 m	2 m
6.3.1	ANISN - Flat ORIGEN source.	53	31	18	9	4	3	4	3	2
6.3.2	ANISN - Ext. ref. - Axial source.	41	25	17	12	6	4	4	3	2
6.3.3	SAS4 - Hom. BU. Flat source	160	36	13	5	1.6	0.9	3.1	0.7	0.2
6.3.4	SAS4 - Real. BU. Flat source	326	70	25	9.6	3.4	1.7	6.5	1.4	0.5
6.3.5	SAS4 - Real. BU. Axial source	67	17	6	16	4.7	2.4	3.3	0.7	0.2
6.3.6	SAS2H - Hom. BU. Flat source	-	-	-	4.5	1.6	0.9	-	-	-

Table 6.3. Neutron doses from safety report -

Fig. 11. Neutron doses at cask wall (radial)

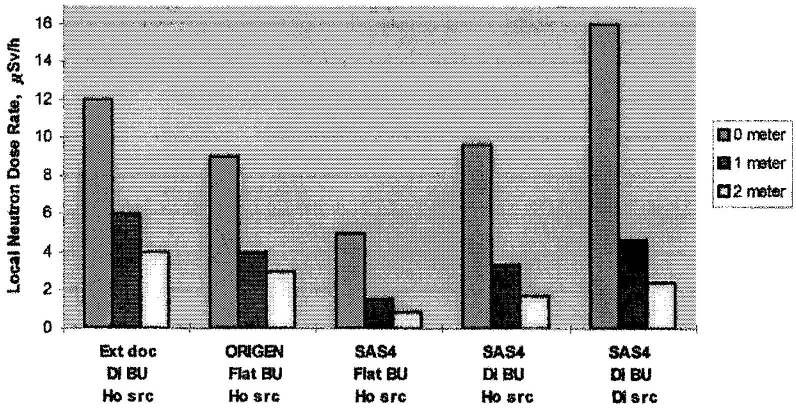


Fig. 12. Neutron doses at cask lid (axial)

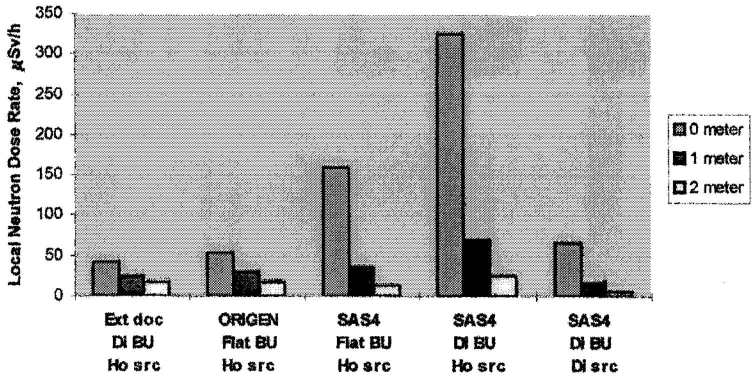
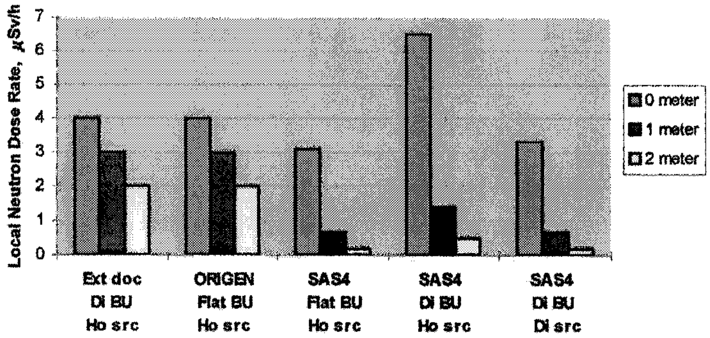


Fig. 13. Neutron doses at cask bottom (axial)



6.3. Observations from comparison of results

6.3.1. Neutron source term as a function of axial burnup distribution

Average axial burnup does not give correct total source terms. The axial burnup distribution is not the same as the axial source distribution, in particular for neutrons.

The safety report uses a safety factor of 1.5 to account for uncertainties in the ORIGEN determination of the neutron source term. The error introduced by assuming a flat burnup distribution appears to be close to a factor 2.0.

6.3.2. Neutron dose variation as a function of distance from centre

Neutron doses obtained with SAS4 are higher at the centre of the axial surface but are reduced fast with increasing radius. Comparing with the safety report, it can be seen that the neutron doses as a function of distance from the axial surface (cask lid or bottom) are reduced much faster with SAS4 than in the safety report.

The SAS4 results seem more correct than the results in the safety report.

6.4. Other comments to the safety report

6.4.1. Using 1-dimensional methods for 2-dimensional problems

Using 1-dimensional methods for dose calculations in the axial direction of a cylindrical cask appears to be difficult. Axial lid doses from a previous safety report for a similar cask are a factor 10 higher, even though the lids are essentially identical. The only difference seems to be the adjustments made to account for the 2-dimensional effect.

6.5. Problems with SCALE 4.3

6.5.1. SAS4 - Point detectors underestimate cask surface doses near symmetry

Many early calculation results indicated that the point detector results were significantly lower than the surface detector results. Even when large surface detectors were used, their average results were often higher than the point detector results at the points of expected maximum doses.

Later, many additional calculations, sometimes with very long running times to reduce the statistical uncertainty, have confirmed that there is indeed a problem with point detectors.

6.5.2. SAS4 - Placing a point detector on or near a radial surface

SAS4 seems to be sensitive to placing a point detector on or close to the radial surface. It was found in many calculations that MORSE generated a tracking error and stopped execution. The cask model had a radius of 117.0 cm. Increasing the radius of the point detector from 117.0 cm by 0.5, 1.0 and 1.5 cm helped in some cases. However, the problem only disappeared when the radius was increased by 2.0 cm to a total of 119.0 cm. The cause of this problem has not been examined.

6.5.3. SAS5 (QADS) - Underestimated doses due to input restriction

As mentioned previously, SAS5 generated much too low gamma doses for the same input as in SAS1 and SAS2H calculations. The restriction to 20 standard compositions in a mixture is not serious if it is known to the user. However, the code system should give an error message rather than carrying out calculations that are not correct. This is particularly important since the results are non-conservative.

6.6. Read the documentation first!

SCALE 4.3 was installed from CD-ROM and downloaded packages from a WWW-site at ORNL. Some documentation was printed out from these sources but older documentation was also used.

A very important lesson learned is to always use the latest documentation. Some changes are extremely important to notice. One example is a change to a standard composition density from the artificial 1.0 to the real theoretical density. Another is the interpretation of MORSE output. The previous documentation /SCALE 4.1) said that uncollided and collided ("total") doses should be added to get the real total dose. The latest documentation says that the total dose now really is the total dose.

Many other delays in the project were caused by lack of or too fast reading of the documentation for SCALE. An excuse for this was that the main purpose of the project was to demonstrate various options in SCALE and to relate them to an independent review of a real safety report. The results and conclusions are not directly related to the safety design or licensing of the cask.

Another source of error is the use of various "working" libraries for cross sections, compositions (ORIGEN) and radiation sources. In one case a series of calculations were carried out with SAS4. Between those calculations some work was also carried out with other sequences. One of those involved a SAS1 sample batch file that copied a special ORIGEN file to the SCALE DATALIB directory and then deleted it from that library. Previously, a file with the same name (FT71F001) had been stored in this library for use with SAS4. This was overwritten. When SAS4 could not find this file in the DATALIB directory it looked in the WORK directory where another file called FT71F001 was found. This was not the one intended.

When there is more than one user of the same computer, the problem with keeping control of the files is even more important. This was the case some of the time during this project. It is not enough to save your own files, it is also necessary to understand what files are used by various sequences in SCALE.

7. Conclusions

The SCALE 4.3 computer code package can be used to independently check safety reports related to gamma and neutron shielding of transport casks. Version 4.4 of SCALE should be released in the near future. It contains significant improvements related to shielding. Allowing more flexible use of surface detectors is one such improvement. Generating colour plots of MORSE input is another expected feature that will be important.

An independent safety review would need more information on the operation of the reactors where the fuel assemblies were irradiated. In addition, information on handling procedures and on the internal structure of the cask with contents is also needed.

There are at least two main differences between the safety report and the results in this project. The first is that the neutron source appears to be stronger than in the safety report. The main reason is that the axial burnup distribution leads to much more higher actinides than if the burnup distribution was flat. Another difference involves the variation of the radiation doses on the surface of the cask and at various distances from the surface. The one-dimensional methods used in the safety report and the rough approximations used to compensate for this limitation probably leads to results that are not realistic. The three-dimensional Monte Carlo method used in the project probably gives more realistic dose distributions.

Missing from the project is an evaluation of the effect of homogenisation of the radioactive contents and internal structure of the cask (steel tubes). How crude this approximation is should be understood before the cask is licensed. This should be combined with further evaluation of a water flooded cask, even though preliminary calculations indicate that this should not be a problem. SCALE can be used for such evaluations but time did not permit them during this project.

8. References

1. Safety report related to the cask design.

2. Various sources of information on the reactors and their operation.

3. SCALE, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-200, Rev. 4, Vol. I-III (April 1995).

4. Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses, O. W. Hermann, S. M. Bowman, M. C. Brady, C. V. Parks, ORNL/TM-12567, (March 1995).

5. Guide to Verification and Validation of the SCALE-4 Radiation Shielding Software, B. L. Broadhead, M. B. Emmett, J. S. Tang. NUREG/CR-6484, (December 1996)

Appendix A - Input data for some typical cases

A selection of some typical input data for various SCALE 4.3 sequences are enclosed.

The input data may or may not produce identical results as in the tables in the report. However, the enclosures are close enough to document typical cases.

Some comments are given below on the composition of the irradiated fuel in SAS4 and QADS calculations. It should be noted that this composition does not influence the neutron and gamma sources, they were generated in SAS2H calculations.

The compositions of the irradiated fuel vary in the examples. The reason is that a single fuel composition was used for each calculation case. Since the real composition of the fuel varies significantly in the axial direction, there is no easy way to specify a single one. A "total composition" could be calculated in the same way as the total neutron and gamma sources were calculated. It was decided that this required too much time for this project.

A change of the burnup from 20 MWd/kgU to 22.5 MWd/kgU changes the fuel enrichment of U-235 significantly. It also changes the total neutron and gamma sources significantly. However, for neutron and gamma shielding, the changed actinide and fission product densities are not important. They influence the radiation self-shielding of the sources and the neutron multiplication. Considering those effects for the cases studied in this project, variations of the irradiated fuel composition are not important.

A.1. SAS2H - Neutron and gamma sources for 20 MWd/kgU

```
=SAS2H      PARM='SKIPCELLWT'
SAS2H  RBMK FUEL: 20 MWD/KGU, 2.0 % U-235, 1 CYC, DRY FUEL CASK
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   MIXTURES OF FUEL-PIN-UNIT-CELL:
'
44GROUPNDF5     LATTICECELL
UO2  1  DEN=10.07  1  840   92234   0.021 92235 2.0 92236 0.011
                         92238 97.968  END
ZIRCONIUM        2  DEN=6.44 0.99 620 END
NB               2  DEN=6.44 0.01 620 END
H2O              3  DEN=0.615  1  547 END
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   MIXTURES OF SHIPPING CASK:
'
'          STAINLESS STEEL TUBES
SS304            4  DEN=0.4366 1.0 350 END
'          SPACERS + CENTRAL TUBES
ZIRCONIUM        4  DEN=0.3254 0.975 350 END
NB               4  DEN=0.3254 0.025 350 END
'          CARBON STEEL IN CASK WALLS AND BASKET
CARBONSTEEL      5  DEN=7.60 1.0 325 END
'          CONCRETE IN CASK - SOME NUCLIDES NOT AVAILABLE
FE               6  DEN=1.6761 1.0 325 END
SILICON          6  DEN=0.0736 1.0 325 END
ALUMINUM         6  DEN=0.0164 1.0 325 END
CALCIUM          6  DEN=0.2532 1.0 325 END
MAGNESIUM        6  DEN=0.083 1.0 325 END
'STRONTIUM        6  DEN=0.0172 1.0 325 END
CARBON           6  DEN=.0605 1.0 325 6012 100 END
SULFUR           6  DEN=.2172 1.0 325 END
'OXYGEN           6  DEN=.8033 1.0 325 8016 99.96 8017 .04 END
OXYGEN           6  DEN=.8033 1.0 325 8016 100.0  END
HYDROGEN         6  DEN=.0223 1.0 325 END
'BA-134           6  DEN=0.8667 .02360 325 END
'BA-135           6  DEN=0.8667 0.06477 325 END
'BA-136           6  DEN=0.8667 0.07769 325 END
'BA-137           6  DEN=0.8667 .11196 325 END
'BA-138           6  DEN=.8667 .72198 325 END
BA-138           6  DEN=.8667 1.0 325 END
'          MATERIAL 0 IN BETWEEN BASKET AND CASK NOT ALLOWED
N               10   DEN=1E-20  1  325 END
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   MIXTURES OF LARGER-UNIT-CELL:
'
'                 ZIRC+NB CENTRAL TUBE
ZIRCONIUM        7  DEN=6.44 0.975 620 END
NB               7  DEN=6.44 0.025 620 END
'                 ZIRC+NB REACTOR CHANNEL
ZIRCONIUM        8  DEN=6.44 0.975 620 END
NB               8  DEN=6.44 0.025 620 END
'                 GRAPHITE - TEMP=750°C.
C-GRAPHITE       9    DEN=1.65 1  1023  END
```

A.2. SAS4 - Lower part, radial gamma case - Distributed source

```
'
END COMP
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'      FUEL-PIN-CELL GEOMETRY:
'
TRIANGPITCH   1.748 1.152  1 3  1.36  2 1.195  0  END
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'    ASSEMBLY AND CYCLE PARAMETERS:
'
NPIN/ASSM=18  FUELNGTH=341  NCYCLES=1  NLIB/CYC=1
PRINTLEVEL=5  LIGHTEL=6 INPLEVEL=2  NUMZONES=6   END
3 0.325 7 0.600 3 0.918  500 4.0 8 4.4 9 14.1
'           ..THESE MIXTURES & RADII PLACE CENTRAL TUBE AT CENTRE OF
'             HOMOGENISED FUEL FOLLOWED BY REACTOR CHANNEL, GRAPHITE
POWER=1.503   BURN=740  DOWN= 1826.25  END
       ZR 28.0   NB  0.5   CR  0.2223   MN  0.0234
       FE  0.8132 NI  0.1112
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   ZONE DESCRIPTION AND OTHER PARAMETERS OF CASK:
'
27N-18COUPLE   TEMPCASK(K)=380 NUMZONES=5 DRYFUEL=YES END
4 62.0 10 73.5  5 78.0  6 113.0  5 117.0
ZONE=1  FUELBNDL=102
  END
END
```

A.3. SAS4 - Lid, neutron case - Distributed source

```
=SAS4      PARM=SIZE=2500000
Concr/Steel cask. RBMK fuel. 2.0% U5. 20 MWd Real BU distr. Low Rad G
27N-18COUPLE    INFHOMMEDIUM
U-234    1  0     5.81E-07         325      END
U-235    1  0     1.66E-05         325      END
U-236    1  0     1.07E-05         325      END
U-238    1  0     3.96E-03         325      END
O        1  0     8.22E-03         325      END
O        3  0     3.03E-02         325      END
ZR       1  0     4.54E-03         325      END
NB       1  0     7.70E-05         325      END
XE-135   1  0     1.83E-19         325      END
CS-133   1  0     5.89E-06         325      END
NP-237   1  0     6.46E-07         325      END
PU-238   1  0     2.01E-07         325      END
PU-239   1  0     1.03E-05         325      END
PU-240   1  0     6.86E-06         325      END
PU-241   1  0     2.22E-06         325      END
PU-242   1  0     1.72E-06         325      END
AM-241   1  0     6.56E-07         325      END
AM-243   1  0     2.31E-07         325      END
CM-244   1  0     3.96E-08         325      END
H        3  0     1.33E-02         325      END
CRSS     1  0     9.61E-04         325      END
MN       1  0     9.57E-05         325      END
FESS     1  0     3.27E-03         325      END
NISS     1  0     4.26E-04         325      END
FE       2  0     8.11E-02         325      END
FE       3  0     1.81E-02         325      END
C        2  0     3.81E-03         325      END
C        3  0     3.04E-03         325      END
SI       3  0     1.58E-03         325      END
AL       3  0     3.66E-04         325      END
CA       3  0     3.80E-03         325      END
MG       3  0     2.06E-03         325      END
S        3  0     4.08E-03         325      END
D        3  0     2.00E-06         325      END
BA-138   3  0     3.78E-03         325      END
END COMP
IDR=0 ITY=2 IZM=5   MHW=0 FRD=62.0   END
62.0 73.5 78.0 113.0 117.0   END
1 0 2 3 2  END
XEND
TIM=240.0 NST=1000 NMT=2000 NIT=20000 NOD=1  SFA=3.90E+16
IGO=0  ISP=2 IPF=5 FR1=0.01 FR2=0.007 FR3=0.007 FR4=0.007  END
DET   119.0 0.0    0.0   END
BUB  0.0    28.4    85.2  142.1    170.5    END
BUF  0.99    1.05    1.01    0.95     0.0    END
GEND
Concrete/Steel cask RBMK fuel - 2.0 WT-% - RADIAL GAMMA - Lower part
FUE  170.5 170.501    END
FEND
CAV  0   73.5  172.5      END
INN  2   78.0  181.5      END
RS1  3 113.0  181.5      END
AS1  3   78.0  216.5      END
HOL  3                    END
OUR  2 117.0  220.5      END
CEND
END
```

```
=SAS4
Concrete/Steel. RBMK fuel. 2.0% U5. Distributed source. Upper Ax N
27N-18COUPLE     INFHOMMEDIUM
U-234    1  0     5.81E-07        325      END
U-235    1  0     1.66E-05        325      END
U-236    1  0     1.07E-05        325      END
U-238    1  0     3.96E-03        325      END
O        1  0     8.22E-03        325      END
O        3  0     3.03E-02        325      END
ZR       1  0     4.54E-03        325      END
NB       1  0     7.70E-05        325      END
XE-135   1  0     1.83E-19        325      END
CS-133   1  0     5.89E-06        325      END
NP-237   1  0     6.46E-07        325      END
PU-238   1  0     2.01E-07        325      END
PU-239   1  0     1.03E-05        325      END
PU-240   1  0     6.86E-06        325      END
PU-241   1  0     2.22E-06        325      END
PU-242   1  0     1.72E-06        325      END
AM-241   1  0     6.56E-07        325      END
AM-243   1  0     2.31E-07        325      END
CM-244   1  0     3.96E-08        325      END
H        3  0     1.33E-02        325      END
CRSS     1  0     9.61E-04        325      END
MN       1  0     9.57E-05        325      END
FESS     1  0     3.27E-03        325      END
NISS     1  0     4.26E-04        325      END
FE       2  0     8.11E-02        325      END
FE       3  0     1.81E-02        325      END
C        2  0     3.81E-03        325      END
C        3  0     3.04E-03        325      END
SI       3  0     1.58E-03        325      END
AL       3  0     3.66E-04        325      END
CA       3  0     3.80E-03        325      END
MG       3  0     2.06E-03        325      END
S        3  0     4.08E-03        325      END
D        3  0     2.00E-06        325      END
BA-138   3  0     3.78E-03        325      END
END COMP
IDR=1 ITY=1 IZM=5   MHW=0 FRD=62.0  END
170.5 203.5 207.5 232.5 236.5  END
1 0 2 2 2  END
XEND
TIM=60.0 NST=1000 NMT=2000 NIT=1000 NOD=1  SFA=4.35E+8
IGO=0  IPF=5 ISP=2  FR1=0.1 FR2=0.1 FR3=0.1 FR4=0.1  END
DET   0.0 0.0 236.5   END
BUB   0.0   28.4   85.2   142.1   170.5   END
BUF   2.13   1.95   0.92    0.13    0.0   END
GEND
Concrete/Steel cask. RBMK fuel. 2.0% U5 - Axial neutrons
FUE  170.5 189.0   END
FEND
CAV  0   73.5  203.499   END
INN  2   78.0  207.5     END
RS1  3 113.0  203.5     END
AS1  2   78.0  228.5     END
HOL  2                  END
OUR  2 117.0  236.5     END
CEND
END
```

A.4. QADS - Modified input to reduce number of nuclides in mixture 1

```
=QADS
Concrete/Steel Cask - RBMK fuel - 2.0 wt-% U-235
ORIGENGP-SRC     INFHOMMEDIUM
U-234    1 0     5.76898E-07   END
U-235    1 0     1.59132E-05   END
U-236    1 0     1.10046E-05   END
U-238    1 0     3.96374E-03   END
O        1 0     8.22049E-03   END
ZR       1 0     4.88421E-03   END
NB       1 0     8.04005E-05   END
'XE-135   1 0     1.82980E-19   END
'CS-133   1 0     5.95607E-06   END
NP-237   1 0     6.66540E-07   END
PU-238   1 0     2.20752E-07   END
PU-239   1 0     9.72621E-06   END
PU-240   1 0     7.11774E-06   END
PU-241   1 0     2.37396E-06   END
PU-242   1 0     1.79232E-06   END
AM-241   1 0     6.98787E-07   END
AM-243   1 0     2.37594E-07   END
CM-244   1 0     3.78628E-08   END
CRSS     1 0     9.60772E-04   END
MN       1 0     9.57174E-05   END
FESS     1 0     3.27218E-03   END
NISS     1 0     4.25613E-04   END
FE       2 0     8.11367E-02   END
C        2 0     3.81402E-03   END
O        3 0     3.02527E-02   END
H        3 0     1.33214E-02   END
FE       3 0     1.80746E-02   END
C        3 0     3.03616E-03   END
SI       3 0     1.57815E-03   END
AL       3 0     3.66036E-04   END
CA       3 0     3.80438E-03   END
MG       3 0     2.05651E-03   END
S        3 0     4.07942E-03   END
D        3 0     1.99881E-06   END
BA-138   3 0     2.73263E-03   END
END COMP
CYLINDRICAL 3.42+16 FLATS  END
NSO=71 NPS=2
END SOURCE
Concrete/Steel CASK - U235 - 2.0 % - QADS
RCC 1 0 0 -170.5 0 0 341.0  62.0
RCC 2 0 0 -172.5 0 0 376.0  73.5
RCC 3 0 0 -181.5 0 0 385.0  78.0
RCC 4 0 0 -216.5 0 0 420.0 113.0
RCC 5 0 0 -220.5 0 0 424.0 117.0
RCC 6 0 0  203.5 0 0  33.0 117.0
RCC 7 0 0  -10000 0 0 20000 20000
RCC 8 0 0  -10001 0 0 20002 20002
END BODY
FUE 2 1
CAV 2 2 -1
INN 2 3 -2
CON 2 4 -3
OUT 2 5 -4
TOP 2 6
BIN 2 7 -5 -6
```

These two nuclides were removed to limit the total number of nuclides to 20

```
BOU 2 8 -7
END ZONE
1 0 2 3 2 2 1000 0
END GEOM
IRON EXP
NDETEC=3
117.0  0.0 0.0  0.0 236.5 0.0 0.0 -220.5 0.0
END DOSE
END
```