SKI Report 02:51

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Nuclear Criticality Safety Assessment Using the SCALE Computer Code Package

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.

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 nuclear criticality safety aspects while a separate report covers radiation shielding.

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, leading to 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. It has been known for many years that criticality safety is very complicated and that independent reviews are absolutely necessary to reduce the risk from quite common errors in the safety assessments.

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. 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.

The safety report appears to lead to correct conclusions. The differences in results are probably caused by somewhat different geometry models. The safety report claims that significant fuel damage is not credible. This needs to be confirmed.

Acknowledgements

It was a pleasure to work with Mr, Arturas Smaizys from the Lithuanian Energy Institute (LEI), In spite of his previous limited experience with SCALE and criticality safety, 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 nuclear criticality safety as well as on radiation shielding.

Very much appreciated during the project was the support from Mr. Curt Bergman from the Swedish International Project (SIP), both in discussions on different subjects, 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

Nuclear criticality safety 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 the nuclear criticality safety 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 shielding.

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.

2. Specifications related to criticality safety

2.1. Safety criteria

No formal criteria for criticality safety are assumed. For this reason it is useful to compare with criteria used internationally.

In some countries, at least two contingencies (unlikely, independent and simultaneous events) must occur before the neutron multiplication k_{eff} can exceed 0.95.

In other countries criticality safety must be assured even after two contingencies. However, k_{eff} may exceed 0.95, but not 0.98, for such combined events.

In both cases, uncertainties and biases in assumptions and methods must be considered.

The interpretation of the contingency concept varies in different countries. Safety in different countries and organisations cannot be compared simply by comparing criteria.

The safety review also involves consideration of the IAEA international transport regulations.

The authority benefits from knowing that incidents 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 the 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 criticality safety control. Unnecessary control, resulting from conservative solutions, requires more resources and may also create worries that have no justification.

2.2. Fissionable material

The concept fissionable material is used here to cover all homogeneous material mixtures that can support a criticality event under certain conditions. The concept fissile material is avoided since it is limited to "thermal neutron" criticality and also because its definition and use in the IAEA transport regulations is confusing.

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.

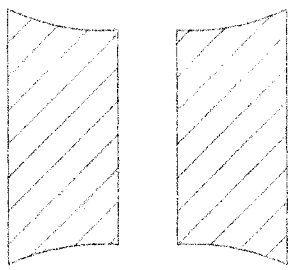


Fig 1. Fuel pellet cut

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. No credit for this absorber is taken in the criticality safety report or review. The average assay of erbium in the fuel pellet is given as between 0.41 and 0.43 wt-%.

For some reactor fuel types, k_{eff} can increase during reactor operation. There is no mention of this possibility in the safety report, maybe because it is not an issue for this reactor type. However, the question should be raised and answered.

In the future, if presence of a burnable absorber (BA) like erbium is taken credit of, the possibility of k_{eff} increase due to irradiation is more likely.

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.

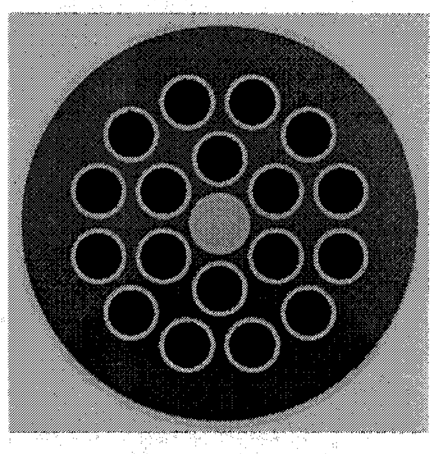


Fig 2. Fuel assembly in steel tube

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

For criticality safety assessment, this information is useful when determining if the neutron multiplication factor can increase as a function of irradiation in the reactor. It will also become important if credit is taken for a reduction in the neutron multiplication factor due to a minimum burnup (burnup credit).

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 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 are simplified. Different models and assumptions are used for criticality safety and shielding assessments.

Radially there is an internal steel basket. This basket is not included in the criticality safety model. The inner carbon steel cylinder of the cask wall has a total thickness of 4.0 cm and an inside diameter of 148 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, in the criticality safety assessment, the fuel is assumed to be in the middle between lid and bottom of the cask. 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 half 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.

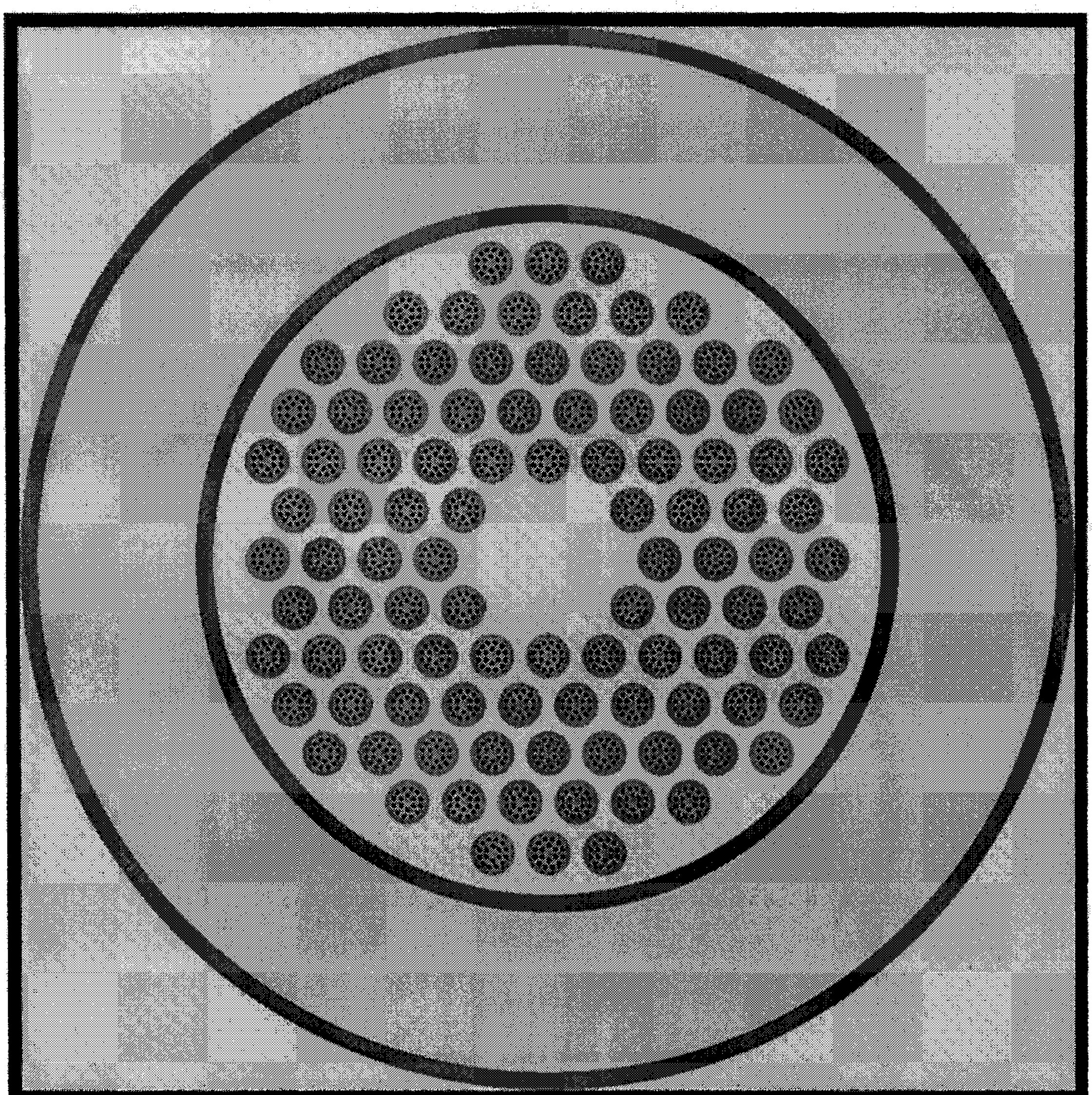


Fig. 3. Cask with half fuel assemblies in steel tubes - Plot from SCALE (KENO-Va)

3. Events related to criticality safety

3.1. Water inside cask

A basic assumption in the safety report is that there can be water inside the cask. This cannot happen during typical 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 cask wall materials may provide more efficient reflection of neutrons than water. For this reason, the presence of a void above the upper fuel level in a cask lying down should be investigated.

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

3.2. Fuel damage

The safety report assumes that there is no damage to the integrity of the fuel or to the fuel assembly. The basis for this assumption is not clear.

The influences of various postulated damages to the geometry of fuel rods and assemblies have been examined in this review. The purpose is not to claim that these damages could be realistic.

The types of fuel damage that have been considered include:

1. Bending rods. It is assumed that a large number of rods can bend in a way that leads to an expansion of the fuel assemblies. It is sufficient if this expansion covers 40 cm or more of the length of the fuel.
2. Water leaks into the gap between clad and fuel pellet and inside the pellets.
3. Fuel rods are broken but the pellets remain inside the clad. The lattice pitch can increase and broken rods may fall into empty positions below. This may lead to an increased fuel density in some axial cross sections of the cask.
4. Fuel pellets fall out of rods and falls to the bottom of the cask.

3.3. Internal structural damage in cask

In the safety report, the fuel assemblies are considered to move inside the stainless steel tubes during or after a drop of the cask. All the movements are considered to be down (gravity).

4. Calculation methods

4.1. SCALE 4.3 as a system

A CD-ROM version of the SCALE 4.3 code package for personal computers was used. 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 was 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. CSAS1X - Neutron multiplication in infinite arrays of rods

The calculation sequence CSAS1X was used in parameter studies. This sequence allows an infinite number of rods in a lattice with a square or triangular pitch. Annular fuel rods can be treated directly. The infinite neutron multiplication factor k_∞ is calculated with the code XSDRNPM-S. Homogenised cross sections can be generated.

4.3. SAS2H - Neutron multiplication in infinite arrays of rods

Reactor burnup calculations were carried out to evaluate the effect of burnup on the infinite neutron multiplication factor, k_∞. SAS2H calculates the neutron spectrum for typical reactor conditions and then applies the depletion and decay code ORIGEN-S for generation of time-dependent nuclide densities. One option is to calculate k_∞ for "cold" reactor conditions. As in CSAS1X, the XSDRNPM-S code is used for that purpose.

4.4. CSAS2X - Complicated geometry with homogenised fuel

The geometry of the cask containing RBMK fuel is quite complicated. In some cases, homogenisation of fuel has been carried out using the SCALE calculation sequence CSAS2X. It includes the same sequence as CSAS1X but adds Monte Carlo techniques to solve problems with the simplification of homogenised fuel lattices.

4.5. CSAS25 - Complicated geometry with a discrete fuel model

In most cases, discrete representation of fuel rods has been used in the SCALE sequence CSAS25. The effective neutron multiplication factor k_{eff} is calculated with the code KENO-Va. Since homogenisation is not used, XSDRNPM-S is not needed.

The sequence CSAS6 based on KENO-VI is available for very complicated geometry. Since KENO-Va is faster, there is no clear advantage at this time of using KENO-VI.

5. Results

5.1. Infinite array of very long fuel pellets in water

This basic case gives information about the fissionable material. Data in criticality safety handbooks can be used to validate the results.

K_∞ was calculated with CSAS1X for infinitely long fuel pellets with varying diameters and in different triangular pitches. The central hole in the pellet and the end zone indentations of the pellets are not directly included. Their effects are represented well enough by the change of diameter and pitch. The UO_2 density is fixed at 10.5 g/cm^3.

Case Id	Pellet diam, cm	Pitch c-c, cm	V_{thH2O}/V_{thUO2}	K_∞
5.1.1	1.00	1.61	1.38	1.35393
5.1.2	1.00	1.80	2.00	1.35323
5.1.3	1.00	1.90	2.35	1.34251
5.1.4	1.00	2.00	2.71	1.32680
5.1.5	1.00	2.10	3.10	1.30720
5.1.6	1.00	2.20	3.50	1.28459
5.1.7	1.15	1.61	1.38	1.31246
5.1.8	1.15	1.80	2.00	1.35362
5.1.9	1.15	1.90	2.35	1.35934
5.1.10	1.15	2.00	2.71	1.35771
5.1.11	1.15	2.10	3.10	1.35028
5.1.12	1.15	2.20	3.50	1.33824
5.1.13	1.30	1.61	1.38	1.22173
5.1.14	1.30	1.80	2.00	1.31471
5.1.15	1.30	1.90	2.35	1.33959
5.1.16	1.30	2.00	2.71	1.35396
5.1.17	1.30	2.10	3.10	1.36013
5.1.18	1.30	2.20	3.50	1.35983

Table 5.1. Pellets in water - varying diameter

The moderation is specified using the theoretical densities of UO_2 and H_2O. The volumes have been modified to account for this. This ratio is similar to the ratio between atomic number densities, H/U in that it describes concentrations and not densities.
This does not mean that the densities and geometry are not important, they are. This can be seen in the results for the same volume ratios but different diameters.

It is important to understand why a change in density or diameter for identical moderation ratios leads to different results. The answer is related to neutron cross section resonances, moderation and absorption in water as well as to fission and absorption in the fuel.

Very small fuel diameters will approach the homogeneous case. Very large diameters lead to very high neutron absorption in water and fuel. There is an optimum in between. This optimum is usually not the same for different conditions. For limited geometry, neutron leakage changes the circumstances. This is also the case with strong neutron absorbers or presence of other types of fissionable material.

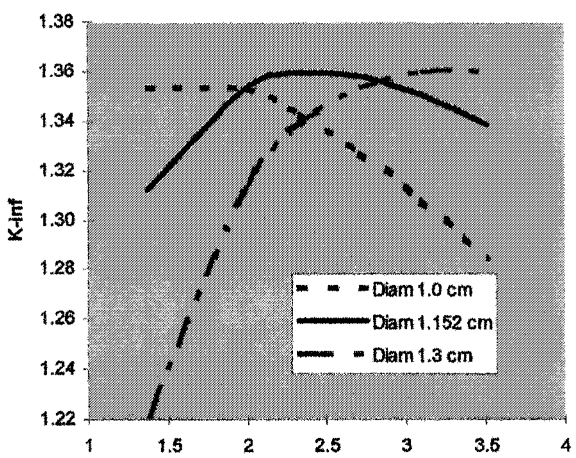


Fig. 4. Moderation - Theoretical volume ratios H2O/UO2

5.2. Infinite array of very long fuel rods in water

This case shows the influence of changed UO_2 densities at varying pitches. The fuel diameter is fixed at 1.152 cm. The outer and inner cladding diameters are fixed at 1.36 cm and 1.195 cm respectively. The pellet end zones and central hole are not directly included, however their effects are covered by the reduced density.

K_∞ was calculated with CSAS1X for infinitely long fuel rods with different UO_2 densities at various triangular pitches.

Case Id	UO2 dens. g/cm3	Pitch c-c, cm	V_{thH2O}/V_{thUO2}	K_∞
5.2.1	10.00	1.605	1.45	1.22217
5.2.2	10.00	1.796	2.09	1.31734
5.2.3	10.00	2.000	2.85	1.34663
5.2.4	10.07	1.605	1.44	1.22119
5.2.5	10.07	1.796	2.07	1.31691
5.2.6	10.07	2.000	2.83	1.34673
5.2.7	10.20	1.605	1.42	1.21936
5.2.8	10.20	1.796	2.05	1.31611
5.2.9	10.20	2.000	2.79	1.34690
5.2.10	10.40	1.605	1.39	1.21656
5.2.11	10.40	1.796	2.01	1.31483
5.2.12	10.40	2.000	2.74	1.34710
5.2.13	10.50	1.605	1.38	1.21517
5.2.14	10.50	1.796	1.99	1.31418
5.2.15	10.50	2.000	2.71	1.34717

Table 5.2. Fuel rods with varying pitch, UO_2 density

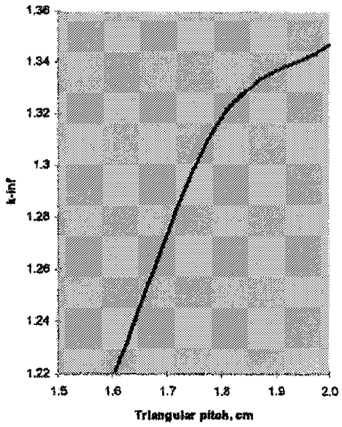


Fig.5. K_∞ versus pitch

5.3. Water leaking into fuel rod gaps and pellet holes

This case involves under-moderated arrays (small pitches).

The annular fuel model in CSAS1X was used to calculate k_∞ for infinitely long fuel rods with different triangular pitches, clad dimensions and central pellet holes. Water is assumed to leak into all fuel rods. The end zone indentations of the pellets are not directly included, nor is the direct influence of increased cladding dimensions, but both effects should be partly covered by varying the central hole diameters in the pellets. The outer fuel pellet diameter is fixed at 1.152 cm. The outer cladding diameter is fixed at 1.36 cm and the inner cladding diameter at the maximum 1.195 cm. The UO_2 density is fixed at 10.5 g/cm^3.

Case Id	Pitch c-c, cm	Centre hole dia cm	Mod V_{thH2O}/V_{thUO2}	K_∞
5.3.1	1.500	0.0	0.92	1.15477
5.3.2	1.500	0.2	0.98	1.17313
5.3.3	1.500	0.4	1.20	1.21915
5.3.4	1.605	0.0	1.25	1.24484
5.3.5	1.605	0.2	1.32	1.25300
5.3.6	1.605	0.4	1.57	1.27589
5.3.7	1.796	0.0	1.90	1.32419
5.3.8	1.796	0.2	1.99	1.32188
5.3.9	1.796	0.4	2.32	1.31898

Table 5.3. Water inside fuel rods

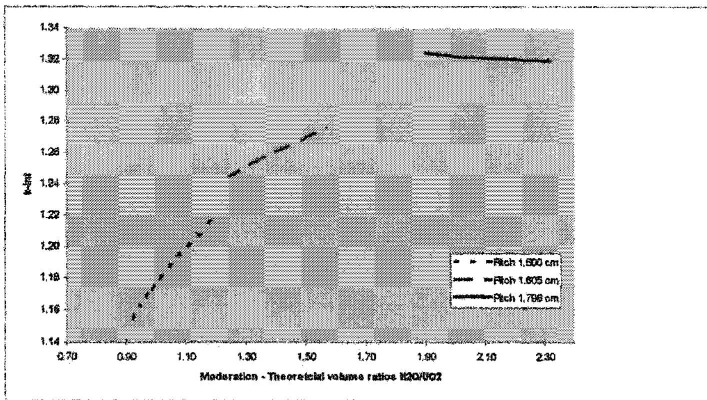


Fig. 6. Centre hole dimension influences

5.4. Influence of various nuclides

If the results show significant negative contributions of a nuclide to the neutron balance, it is important to verify the validity of the cross sections for similar systems. It is also important to verify that the nuclides are really present in the operation being reviewed.

If the nuclides are expected to give a positive contribution to the neutron balance (fission and possibly other reactions increasing the number of neutrons), it is important to verify that the results include such effects adequately.

Neutron scattering can be important both for increasing and reducing the neutron multiplication factor. Scattering reduces neutron leakage from a material. Whether the effect is positive or negative depends on the materials and geometry. In a similar way scattering leads to increased moderation in the material. Again, this can have positive or negative effects on the neutron multiplication factor.

The nuclides checked here are related to the fuel (uranium isotopes) and the cladding (zirconium and niobium). Water has already been tested sufficiently. The influence of 0.41 wt-% erbium mixed with the fuel is estimated.

The reference case has the following specifications. The geometry is a solid, infinite fuel pellet with density 10.07 g/cm^3 and outer diameter 1.152 cm. The cladding has an outer diameter of 1.36 cm and an inner diameter of 1.195 cm. The triangular pitch is 1.796 cm.

Case Id	Material change related to reference case	K_∞
5.4.1	Reference case	1.31691
5.4.2	2.0 wt-% U-235 in uranium	1.26919
5.4.3	2.8 wt-% U-235 in uranium	1.35351
5.4.4	Cladding without Nb	1.31821
5.4.5	Cladding with 2.5 wt-% Nb	1.31526
5.4.6	Cladding without Zr	1.33478
5.4.7	0.021% U-234,0.011% U-236,97.568% U-238	1.31384
5.4.8	0.41 wt-% erbium in UO2	1.07366

Table 5.4. Influence of various nuclides

5.5. Influence of various water densities

It is traditional in criticality safety analysis to evaluate the influence of reduced, but homogeneous water densities. The evaluation will give information on some of the neutron physics parameters of the system. The optimum condition may not be realistic, but if it is safe enough, further safety analysis and implementation could be simplified.

If the fuel is not irradiated, as assumed in the criticality safety report, there will not be any heat from radioactive decay. If there is significant decay heat, the fuel must have been irradiated to a high burnup and the decay (cooling) time short.

Since burnup (depletion) calculations will be made both for criticality safety and for shielding, it is also of interest to calculate the neutron multiplication for various temperatures. Also, the density of water can vary for the same temperature (boiling). However, here only the water density is changed, not the temperature.

The reference case has the following specifications. The geometry is a solid, infinite fuel pellet with density 10.07 g/cm^3 and outer diameter 1.152 cm. The cladding has an outer diameter of 1.36 cm and an inner diameter of 1.195 cm. The triangular pitch is 1.796 cm. The temperatures in water, cladding and fuel is the same, 293 K.

Case Id	Temperatures	H2O dens g/cm3	K_∞
5.5.1	Reference case	0.998	1.31689
5.5.2	All materials 373K, no boiling	0.958	1.31141
5.5.3	All materials 373K, boiling	0.500	1.18065
5.5.4	All materials 552K	0.760	1.25734
5.5.5	Fuel 800K, cl 620K, H2O 552K	0.760	1.24798
5.5.6	Fuel 800K, cl 620K, H2O 530K	0.785	1.25438
5.5.7	Fuel 800K, cl 620K, H2O 552K	0.500	1.15713
5.5.8	Fuel 800K, cl 620K, H2O 552K	0.250	0.97774
5.5.9	Fuel 700K, cl 620K, H2O 552K	0.760	1.25155
5.5.10	Fuel 900K, cl 620K, H2O 552K	0.760	1.24462

Table 5.5. Influence of temperature

5.6. Influence of burnup

The safety report does not propose credit for burnable absorbers or burnup. However, since the fuel and reactor types have not been evaluated by the author before, a quick demonstration of SCALE for this purpose was tried.

K_∞ was calculated with SAS2H for infinitely long fuel rods in water, after being irradiated under different reactor conditions. A SAS2H model description follows.

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 position is assumed to be divided between the 18 fuel rods. That is expected to be 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, giving a diameter of about 28 cm.

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 density, time of reactor operation, time of decay (cooling), temperatures and densities.

As a part of calculating the neutron energy spectrum, SAS2H uses the XSDRNPM-S code for calculation of k_∞. Normally this will be at "hot" reactor operating conditions. To get k_∞ for "cold" reactor conditions, SAS2H can be made to calculate reactor cycles of very short times and power densities at temperatures and densities that are typical for storage or transport conditions. This is a quick way to estimate the influence of burnup on k_∞.

SAS2H automatically inserts trace quantities of actinides and two fission products in the input before depletion calculations start. To get a realistic influence of the burnup and the decay time, trace quantities of other fission products need to be included in the "cold" condition. This is necessary for allowing XSDRNPM-S to include the influence of these nuclides as a function of time.

The specifications for the fuel cell of the burnup calculation are as follows. Fuel pellet diameter 1.152 cm, UO_2 density 10.07 g/cm^3, cladding inner diameter 1.195 cm, cladding outer diameter 1.36 cm and pitch 1.796 cm. The water density during reactor operation is varied as is the burnup. The results are for "cold" fuel at full water density.

The results are shown in table 5.6 and figure 7..

30 D decay, 2.4% U5, Reactor WD=0.24 g/cm3. "Cold" WD=4.16*0.24			
Case Id	Depl. Days	K_∞ No BA	K_∞ Erbium
5.6.1	0	1.33702	1.08303
5.6.2	18.5	1.32031	1.08601
5.6.3	37	1.31307	1.09456
5.6.4	55.5	1.30565	1.10182
5.6.5	74	1.29783	1.10775
5.6.6	92.5	1.29055	1.11323
5.6.7	111	1.28151	1.11714
5.6.8	129.5		1.12025
5.6.9	148	1.26476	1.12286
5.6.10	166.5	1.25595	1.12438
5.6.11	185		1.12582
5.6.12	203.5	1.23963	1.12673
5.6.13	222		1.12712
5.6.14	240.5	1.22356	1.12701
5.6.15	259		1.12642
5.6.16	277.5	1.20763	1.12534
5.6.17	296		1.12375
5.6.18	314.5	1.19176	1.12181
5.6.19	333		1.11948
5.6.20	351.5	1.17602	1.11673
5.6.21	370	1.16816	1.11361
5.6.22	407	1.15246	1.10633
5.6.23	444	1.13674	1.09778
5.6.24	481	1.12103	1.08816
5.6.25	518	1.10531	1.07762
5.6.26	555	1.08957	1.06627
5.6.27	592	1.07384	1.05431
5.6.28	629	1.05813	1.04183
5.6.29	666	1.04245	1.02894
5.6.30	703	1.02682	1.01576
5.6.31	740	1.01127	1.00238

Table 5.6. Influence of burnup

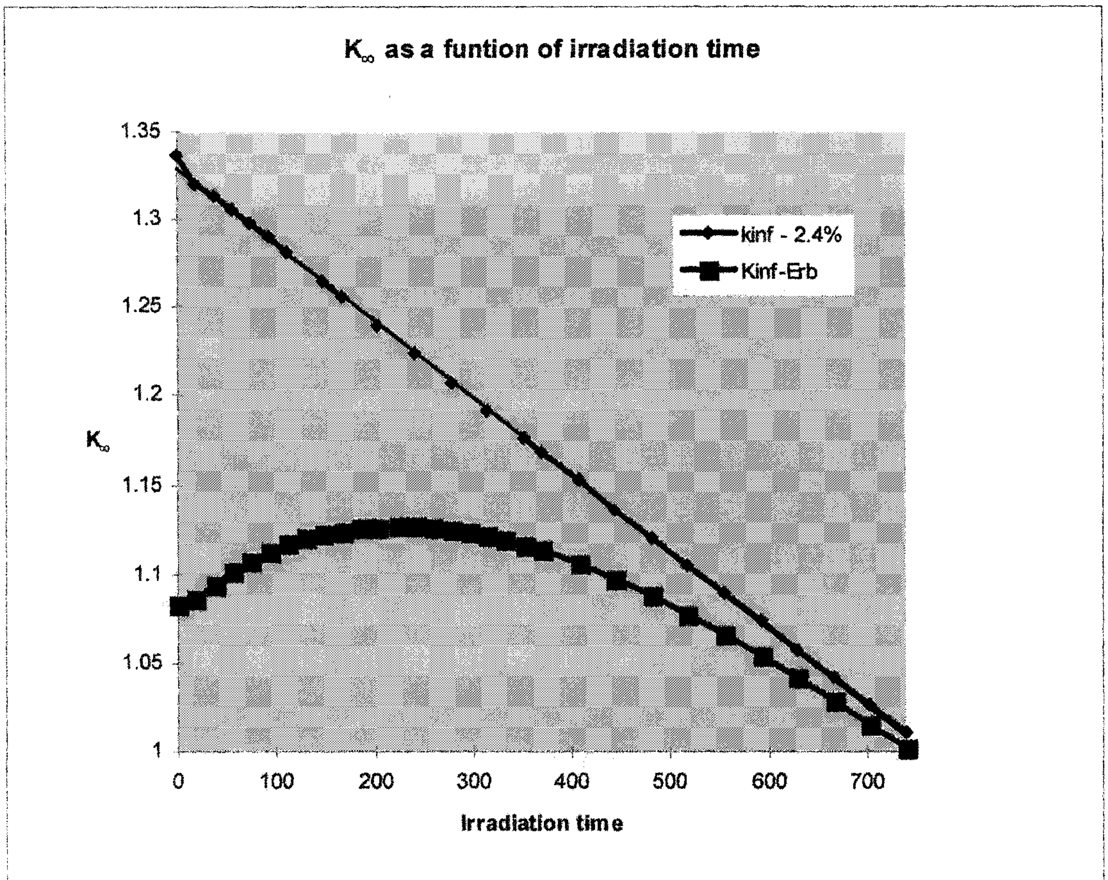


Figure 7. K_∞ as a function of burnup for fuel with and without erbium

5.7. Fuel assembly calculations

The irregular design of the fuel assembly can be described accurately in KENO-Va geometry. However, the calculations of problem-dependent nuclear cross sections as a function of neutron energy and space can only be carried out in simplified geometry.

The pitch (centre-to-centre distance) between fuel rods and the pattern of the rod lattice (square or triangular) are some of the parameters that may be important.

The material surrounding the fuel assembly and the arrangement of assemblies are other parameters of importance. The steel in the tubes surrounding the assemblies is one such material. Air or other materials between the tubes may not be realistic, but calculations of such changes may give important information.

A change of the separation between the storage tubes is a parameter that is considered in the safety report.

Optimisation of k_∞ for fuel rods as a function of various parameters will often not result in the same values as optimisation of k_∞ or k_{eff} for fuel assemblies.

Case Id	KENO-Va. Infinite length assemblies in 12.5 cm centre-to-centre steel tubes, infinite array, water in guide tube	K_∞	σ
5.7.1	1.796 cm triang. pitch in cross section (XS) preparation	1.0472	0.0018
5.7.2	Infinite array of 346 cm fuel assemblies. 20 cm H2O refl.	1.0330	0.0021
5.7.3	1.796 cm triang. pitch (XS). Void in guide tube	1.0430	0.0018
5.7.4	1.605 cm triang. pitch (XS). Void in guide tube	1.0430	0.0018
5.7.5	2.000 cm triang. pitch.(XS). Void in guide tube	1.0359	0.0020
5.7.6	Eccentric fuel assembly (near wall). Periodic boundary c.	1.0448	0.0018
5.7.7	Eccentric fuel assembly (near wall). Reflect. boundary c.	1.0172	0.0016
5.7.8	Steel tube wall thickness reduced from 2.0 cm to 1.8 cm	1.0444	0.0018
5.7.9	Steel tube wall thickness reduced from 2.0 cm to 0.0 cm	1.2540	0.0016
5.7.10	Homogenised fuel (1.796 cm pitch). Fuel diameter 8.0 cm	1.0731	0.0015
5.7.11	Homogenised fuel (1.796 cm pitch). Fuel diameter 7.8 cm	1.0531	0.0016

Table 5.7. Fuel assembly in steel tube

5.8. Cask geometry - "Normal" conditions with water

Some calculations were carried out for fuel rods that are homogenised with water using the CSAS2X sequence.

The final conclusions should be verified by calculations using realistic geometry. Use of discrete representation of fuel rods is recommended for such cases.

Case Id	KENO-Va.	K_∞	σ
5.8.1	Normal cask - No water inside	0.2396	0.0008
5.8.2	Normal cask - Water inside (in all remaining cases)	0.7229	0.0017
5.8.3	Normal cask - As case 5.8.2 except: No steel tubes	0.8585	0.0028
5.8.4	Normal cask - As case 5.8.2 except: Void between tubes	0.9583	0.0021

Table 5.8. Normal cask with water

5.9. Cask geometry - Accident conditions with water

Case Id	KENO-Va. Tube structure collapsed. All tubes in contact in triangular lattice.	K_∞	σ
5.9.1	All tubes moved towards centre of cask	0.9367	0.0020
5.9.2	Horizontal cask - All tubes on bottom - flat surface	0.9362	0.0021
5.9.3	Horizontal cask - Void above fuel level	0.9335	0.0020
5.9.4	Horizontal cask - Eccentric assemblies	0.9347	0.0019
5.9.5	Hom. fuel. Rod pitch=1.796 cm. Fuel diam=8.0 cm	0.9592	0.0018
5.9.6	Hom. fuel. Rod pitch=2.020 cm. Fuel diam=9.0 cm	1.0244	0.0018
5.9.7	Hom. fuel. Rod pitch=2.200 cm. Fuel diam=9.8 cm	1.0505	0.0017
5.9.8	As case 5.6.5 but void outside tubes	0.9680	0.0017
5.9.9	As case 5.6.5 but water instead of steel/concrete wall	0.8886	0.0022

Table 5.9. Damaged cask

5.10. Normal and accident cask geometry - Low-density water

A traditional criticality safety assessment must include calculations on a homogeneous, uniform distribution of low density water in all available spaces even if there are no credible scenarios leading to such a situation. The probability for these scenarios are not evaluated.

Case Id	H2O g/cm3	K_{eff} Normal	σ Normal	K_{eff} Accident	σ Accident
5.10.1	0.20	0.8101	0.0017	0.7544	0.0018
5.10.2	0.30	0.8404	0.0018	0.8386	0.0018
5.10.3	0.35	0.8439	0.0021		
5.10.4	0.40	0.8447	0.0018	0.8926	0.0018
5.10.5	0.45	0.8399	0.0019		
5.10.6	0.50	0.8312	0.0018	0.9209	0.0021
5.10.7	0.60	0.8152	0.0021	0.9411	0.0006
5.10.8	0.70	0.7959	0.0019	0.9454	0.0011
5.10.9	0.80	0.7719	0.0019	0.9470	0.0011
5.10.10	0.90	0.7443	0.0020	0.9423	0.0012
5.10.11	1.00	0.7229	0.0017	0.9296	0.0019

Table 5.10. Varying water density

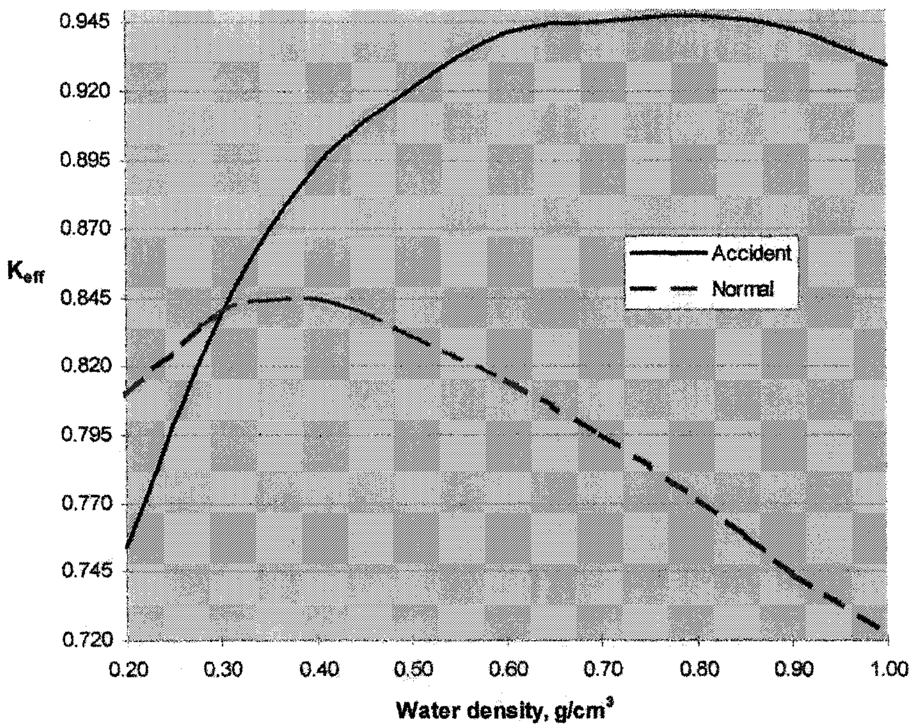


Figure 7. K_{∞} as a function of burnup for fuel with and without erbium

Accident cases for water densities 0.6 to 0.9 g/cm³ were recalculated (increasing total neutrons from 100 000 to 300 000) since the statistical trend lines shown by KENO-Va indicated non-convergence. It is likely that many other cases would also give results outside of the current 2*σ level if rerun with better statistics. Time did not allow more accurate calculations. Trend lines from these cases are shown in chapter 6.

6. Discussion of results

6.1. Influence of reactor irradiation (burnup) on criticality safety

The results of the burnup calculations indicate that there is no combination of parameters for fuel without burnable absorbers (erbium) that would give a higher k_{eff} with increasing burnup. With erbium there appears to be such an effect. This conclusion is in agreement with experience from BWR reactors.

If the erbium is not taken credit for, the assumption of unirradiated fuel is a conservative approach.

6.2. Fuel damage

The results from the calculations of various fuel rod pitches and of the influence of the steel tubes can be used to draw conclusion on potential damage to the fuel.

If the fuel rods are bent outwards, leading to an expansion of each assembly along a significant part (probably more than 40 cm) of its length, k_{eff} can increase above the accepted limit.

If parts of the fuel assemblies, as a consequence of an accident, are positioned outside the steel tubes, k_{eff} can increase significantly.

If the fuel rods can break so that pellets or broken pellets can be distributed anywhere in the cask, k_{eff} could become unacceptably high.

6.3. Structural damage to other parts than fuel

The results from the calculations of eccentric fuel assemblies inside the steel tubes indicate that there is no substantial increase in k_{eff}. If the steel tubes were closed and contained some water, the conclusion would be different. However, that does not seem to be a possible scenario.

If the internal structure of steel tubes collapses so that the tubes become in contact with each other in a triangular array, k_{eff} will increase but the cask will stay sub-critical.

6.4. Statistical considerations

During the project, the statistical basis of Monte Carlo Codes (KENO-Va and MORSE) was accounted for in different ways. A general principle that is important in this type of calculation is that the result should be known in advance of the calculation. This puts a limit to how big a step should be taken from previous validated calculations or experiments. As soon as a result is outside of the expected range, it is important to study the reasons. Input errors, violation of code limitations and statistical behaviour explain many unexpected results. Bugs in the computer software and errors in the data libraries should not be ruled out.

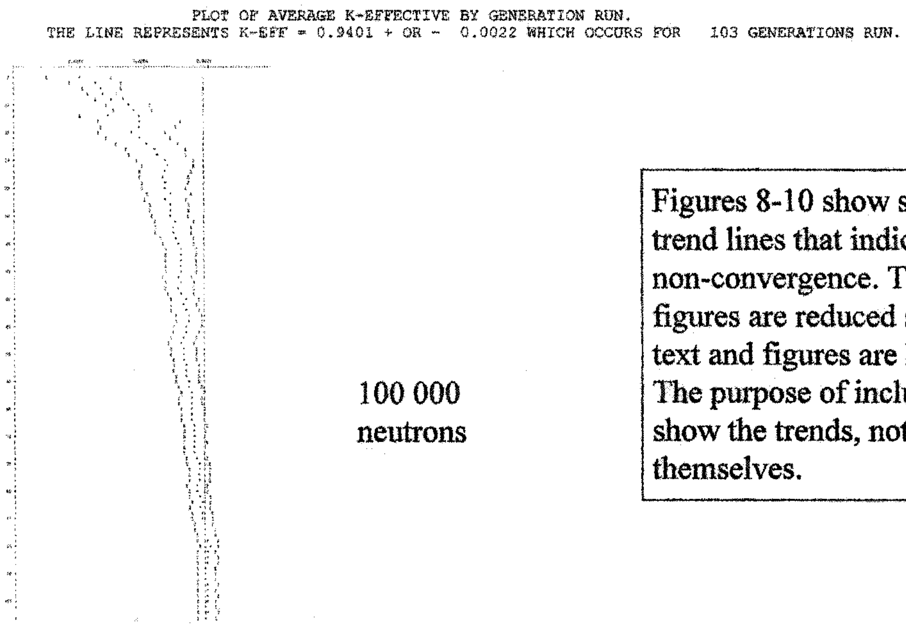
Two ways of check the statistics of a completed calculation is to look at the trend lines and to verify that the neutron fission distribution is close to what was expected.

Time did not allow very accurate Monte Carlo calculations. In most cases only 100 000 neutrons were run. The trend lines indicated that for some of the cases, the correct results may not have been found. In the study of varying the water density in the cask, some cases were recalculated using better statistics (300 000 and 1 000 000 neutrons).

Experience with statistical complications with Monte Carlo supports the addition of three rather than two standard deviations to the calculated k_{eff}. However, the standard deviation is not the proper way to deal with deviations that are explained by slow convergence or convergence to the wrong solution. The later effect can be seen when there are several geometry regions "competing for the attention" of the neutrons.

Figure 8 shows a trend line that obviously indicated that more neutrons were required to get a reliable result. It was generated for the accident case with water at a density of 0.9 g/cm^3 while figures 9 and 10 were generated for a water density of 0.9 g/cm^3.

Figures 9 and 10 were generated by the same KENO-Va calculation. Figure 9 shows k_{eff} as a function of adding more generations. The total statistics improve going down the curve. However, a change of direction near the end (could be that the neutrons finally approach the hottest region) may be drowned by all the previous statistics. Figure 10 shows k_{eff} as a function of removing initial batches of neutrons. The total statistics are reduced going down the curve. However, late trends are often shown in a clear way.



100 000
neutrons

Figures 8-10 show some typical trend lines that indicate non-convergence. The sizes of the figures are reduced so much that the text and figures are hard to read. The purpose of including them is to show the trends, not the numbers in themselves.

Fig. 8. A trend line indicating a possible underestimation of k_{eff}

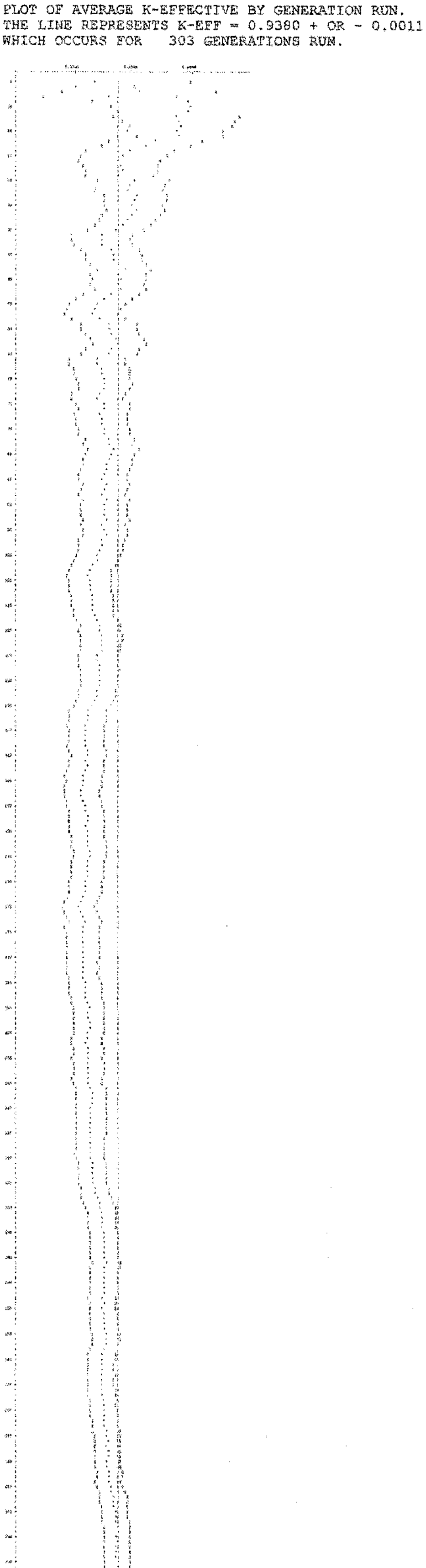


Fig. 9. A trend line changing direction

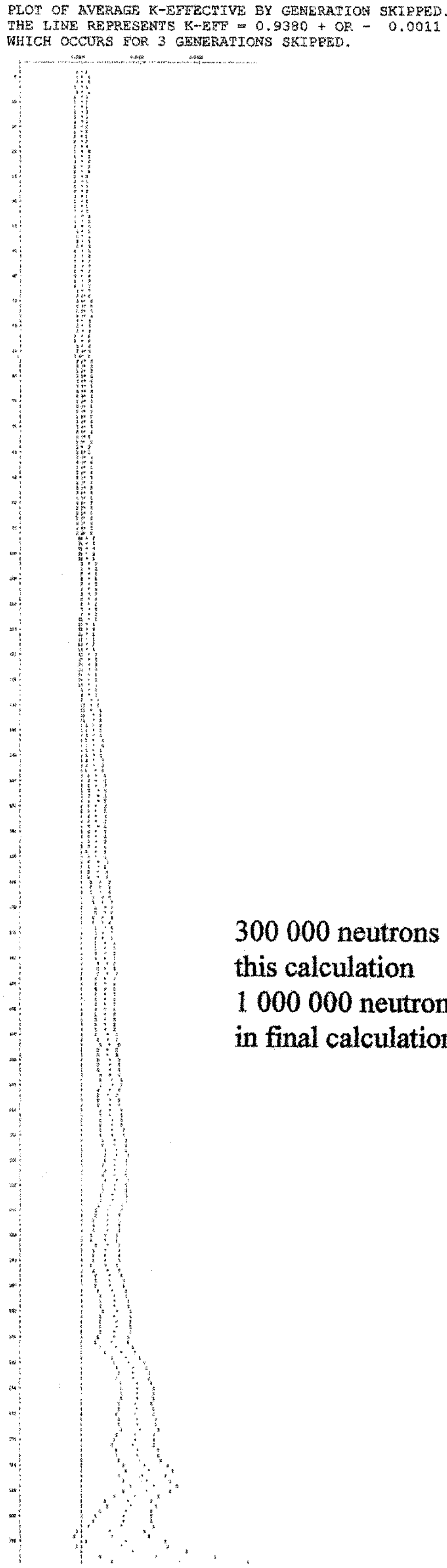


Fig. 10. A different trend line for the same case

300 000 neutrons in this calculation
1 000 000 neutrons in final calculation

7. Comparison with safety report for the cask

The safety report includes a criticality safety assessment based on calculations with SCALE 4.3. The same codes and cross section library as in this project were used.

The results in the safety report basically agree with those that were obtained during this project. Two differences should be noted:

1. The exact configuration of the damaged case is not described in the safety report. The triangular array of steel tubes in contact can be arranged in different ways. A difference here probably explains why the peak k_{eff} is a little bit lower in the safety report.
2. In accident cases, the safety report uses a steel wall thickness of 1.8 mm in all tubes. During this project, the nominal thickness of 2.0 mm was used. This difference makes k_{eff} higher in the safety report.

The two differences probably influence k_{eff} in different directions, making the total calculated difference smaller.

Considering that the same calculation method was used in the safety report as in this project, the large differences for the normal cases are somewhat surprising. Concerning the large differences in the low water density accident cases, the results may be explained by the difference in steel tube wall thickness.

8. Conclusions

The SCALE 4.3 computer code package can be used to independently check safety reports related to nuclear criticality safety of transport casks. Important limitations, that are still present in SCALE, involve cross section processing and resonance treatment in particular. Varying pitches, different geometry or compositions in different fuel rods and other two-dimensional effects cannot be treated easily. Treatment of overlapping resonances (different nuclides having resonances that are close to each other in energy) is another weakness. A future version of SCALE (version 5) will include some improvements on this.

Concerning the safety of the storage cask, the assessment appears quite easy. As long s the cask is internally dry there is no potential for criticality. If one or more casks are filled with water, the safety appears to be assured as long as there is no significant damage to a large part of the fuel in the cask.

There are some differences between calculated results according to the safety report and those according to this project. They are not significant for the conclusions concerning the safety of the cask and they can probably be explained easily if more information about the safety report calculations is obtained.

9. 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 Criticality Safety Software*, M. B. Emmett, W. C. Jordan. NUREG/CR-6483, (December 1996)

6. *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,* J. J. Lichtenwalter, S. M. Bowman, M. D. DeHart, C. M. Hopper, NUREG/CR-6361, (March 1997)

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.

A.1. CSAS1X - Infinite lattice of fuel rods

```
=CSAS1X
RBMK FUEL PELLET
44GROUPNDF5                  LATTICECELL
UO2          1  DEN=10.070 1.0 293 92235 2.4 92238 97.6 END
ZIRCONIUM    2  DEN=6.44 0.99 293 END
NB           2  DEN=6.44 0.01 293 END
H2O          3  1.0 293 END
END COMP
TRIANGPITCH 1.796 1.152 1  3  1.36 2  1.195 0  END
END
```

A.2. CSAS1X - Fuel rods with annular pellets and water inside rods

```
=CSAS1X
RBMK FUEL PELLET
44GROUPNDF5                  LATTICECELL
UO2            1  DEN=10.5 1.0 293 92235 2.4 92238 97.6 END
ZIRCONIUM      2  DEN=6.44 0.99 293 END
NB             2  DEN=6.44 0.01 293 END
H2O            3  1.0 293 END
H2O            4  1.0 293 END
H2O            5  1.0 293 END
END COMP
ATRIANGPITCH  1.605 1.152 1 3 4 0.2  1.36  2  1.195  5  END
END
```

A.3. SAS2H - Burnup credit with fuel containing erbium

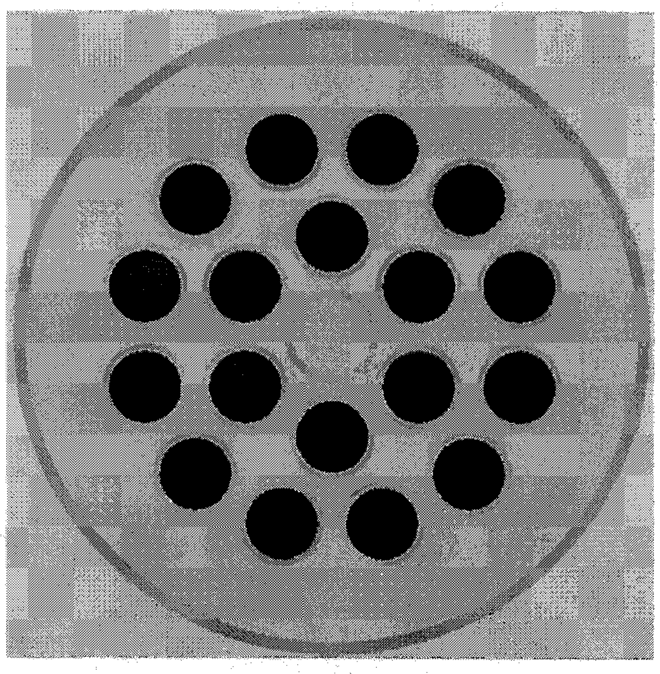
```
=SAS2H      PARM='SKIPCELLWT,SKIPSHIPDATA'
SAS2H TEST RBMK: 20 MWD/KGU, 2.4 % U-235, 1 CYC, DRY FUEL CASK
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   MIXTURES OF FUEL-PIN-UNIT-CELL:
'
44GROUPNDF5     LATTICECELL
UO2  1  DEN=10.07  0.9959 840  92234  0.021 92235 2.4 92236 0.011
                   92238 97.568  END
ERBIUM          1  DEN=10.07 0.0041 293 68166 60 68167 40 END
PU-238          1  0 1E-20  840  END
PU-239          1  0 1E-20  840  END
PU-240          1  0 1E-20  840  END
PU-241          1  0 1E-20  840  END
PU-242          1  0 1E-20  840  END
AM-241          1  0 1E-20  840  END
AM-243          1  0 1E-20  840  END
NP-237          1  0 1E-20  840  END
MO-95           1  0 1E-20  840  END
TC-99           1  0 1E-20  840  END
RU-101          1  0 1E-20  840  END
RU-103          1  0 1E-20  840  END
AG-109          1  0 1E-20  840  END
CS-133          1  0 1E-20  840  END
CS-135          1  0 1E-20  840  END
SM-147          1  0 1E-20  840  END
SM-149          1  0 1E-20  840  END
SM-150          1  0 1E-20  840  END
SM-151          1  0 1E-20  840  END
SM-152          1  0 1E-20  840  END
ND-143          1  0 1E-20  840  END
ND-145          1  0 1E-20  840  END
EU-153          1  0 1E-20  840  END
GD-155          1  0 1E-20  840  END
ZIRCONIUM       2  DEN=6.44 0.99 620 END
NB              2  DEN=6.44 0.01 620 END
H2O  3 DEN=0.24  1  552 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 1.0 293 END
FE              6  DEN=1.6761 1.0 293 END
'           CONCRETE IN CASK
SILICON         6  DEN=0.0736 1.0 293 END
ALUMINUM        6  DEN=0.0164 1.0 293 END
CALCIUM         6  DEN=0.2532 1.0 293 END
MAGNESIUM       6  DEN=0.083 1.0 293 END
'STRONTIUM       6  DEN=0.0172 1.0 293 END
CARBON          6  DEN=.0605 1.0 293 6012 100 END
SULFUR          6  DEN=.2172 1.0 293 END
```

The additional fission products that are included are those that were selected as important by an OECD Working Group on burnup credit. The densities will be 10^{-20} at start with fresh fuel, but will increase with increasing burnup.

```
'OXYGEN          6  DEN=.8033 1.0 293 8016 99.96 8017 .04 END
OXYGEN          6  DEN=.8033 1.0 293 8016 100.0  END
HYDROGEN        6  DEN=.0223 1.0 293 END
'BA-134          6  DEN=0.8667 .02360 293 END
'BA-135          6  DEN=0.8667 0.06477 293 END
'BA-136          6  DEN=0.8667 0.07769 293 END
'BA-137          6  DEN=0.8667 .11196 293 END
BA-138          6  DEN=.8667 .72198 293 END
'         ERROR IF MATERIAL 0 IN BETWEEN BASKET AND CASK
N    10   DEN=1E-20  1  293 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
'
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=3  NLIB/CYC=1
PRINTLEVEL=1  LIGHTEL=0 INPLEVEL=2  NUMZONES=6   END
3 0.325 7 0.600 3 0.943  500 4.0 8 4.4 9 14.1
'          ..THESE MIXTURES & RADII PLACE CENTRAL TUBE AT CENTER
'            HOMOGENIZED FUEL, REACTOR CHANNEL, GRAPHITE
POWER=1.503   BURN=222.    DOWN= 0.0                            END
POWER=1.0E-6  BURN=1.0E-6  DOWN=30.0   H2OFRAC=4.16 TEMKCYC=293  END
POWER=1.0E-6  BURN=1.0E-6  DOWN=0.0   H2OFRAC=4.16 TEMKCYC=293  END
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   ZONE DESCRIPTION AND OTHER PARAMETERS OF CASK:
'
27N-18COUPLE   TEMPCASK(K)=380 NUMZONES=5 DRYFUEL=YES END
4 62.0 10 74.0  5 78.0  6 113.0  5 117.0
ZONE=1  FUELBNDL=102
  END
END
```

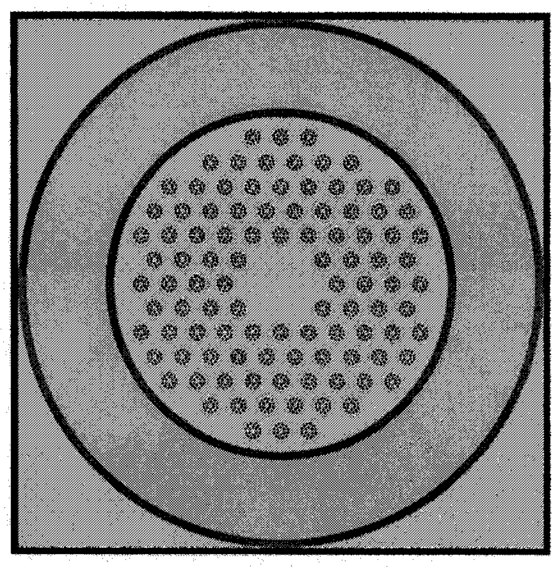
A.4. CSAS25 - Fuel assembly with central rod

```
=CSAS25
RBMK FUEL RODS
44GROUPNDF5                  LATTICECELL
UO2              1  DEN=10.07 1.0 293 92235 2.4 92238 97.6 END
ZIRCONIUM        2  DEN=6.44 0.99 293 END
NB               2  DEN=6.44 0.01 293 END
H2O              3  1.0 293 END
SS304            4  DEN=7.60 1.0 293 END
H2O              5  1.0 293 END
END COMP
TRIANGPITCH  1.796 1.152 1  3  1.36 2  1.195 0  END
Assembly with guide tube (PITCH=1.796 TRI)
READ PARM RUN=YES PLT=NO GEN=103 NPG=1000 NUB=YES FDN=YES END PARM
READ GEOM
UNIT 1
COM=!FUEL CELL!
CYLINDER   1 1 .576  173 -173
CYLINDER   0 1 .5975 173 -173
CYLINDER   2 1 .68   173 -173
UNIT 2
COM=!ROD IN CENTER!
CYLINDER   3 1 0.625 173 -173
CYLINDER   2 1 0.75  173 -173
GLOBAL UNIT  3
COM=!Tube + rods!
CYLINDER   5 1 4.9    173 -173
HOLE  1 0.0 1.6 0.0
HOLE  1 0.0 -1.6 0.0
HOLE  1 1.386 0.8 0.0
HOLE  1 1.386 -0.8 0.0
HOLE  1 -1.386 -0.8 0.0
HOLE  1 -1.386 0.8 0.0
HOLE  1 2.994 0.802 0.0
HOLE  1 2.994 -0.802 0.0
HOLE  1 -2.994 -0.802 0.0
HOLE  1 -2.994 0.802 0.0
HOLE  1 2.192 2.192 0.0
HOLE  1 2.192 -2.192 0.0
HOLE  1 -2.192 -2.192 0.0
HOLE  1 -2.192 2.192 0.0
HOLE  1 0.802 2.994 0.0
HOLE  1 0.802 -2.994 0.0
HOLE  1 -0.802 -2.994 0.0
HOLE  1 -0.802 2.994 0.0
HOLE  2  0.0 0.0 0.0
CYLINDER   4 1 5.1 173 -173
CUBOID     0 1 6.1 -6.1 6.1
-6.1 173 -173
END GEOM
READ BNDS +XB=REFLECT
-XB=REFLECT +YB=REFLECT
-YB=REFLECT +ZB=REFLECT
 -ZB=REFLECT  END BNDS
END DATA
END
```



A.5. CSAS25 - Dry cask with discrete model of fuel rods

```
=CSAS25
RBMK FUEL RODS
44GROUPNDF5                LATTICECELL
UO2             1  DEN=10.07 1.0 293 92235 2.4 92238 97.6 END
ZIRCONIUM       2  DEN=6.44 0.99 293 END
NB              2  DEN=6.44 0.01 293 END
H2O             3  1.0 293 END
SS304           4  DEN=7.60 1.0 293 END
FE              5  DEN=1.6761 1.0 293 END
SILICON         5  DEN=0.0736 1.0 293 END
ALUMINUM        5  DEN=0.0164 1.0 293 END
CALCIUM         5  DEN=0.2532 1.0 293 END
MAGNESIUM       5  DEN=0.083 1.0 293 END
STRONTIUM       5  DEN=0.0172 1.0 293 END
BA-134          5  DEN=0.8667 .02360 293 END
CARBON          5  DEN=.0605 1.0 293 6012 100 END
SULFUR          5  DEN=.2172 1.0 293 END
OXYGEN          5  DEN=.8033 1.0 293 8016 99.96 8017 .04 END
HYDROGEN        5  DEN=.0223 1.0 293 END
CARBONSTEEL     6  DEN=7.60 1.0 293 END
IRON            7  DEN=7.60 1.0 293 END
H2O             8  1.0 293 END
BA-135          5  DEN=0.8667 0.06477 293 END
BA-136          5  DEN=0.8667 0.07769 293 END
BA-137          5  DEN=0.8667 .11196 293 END
BA-138          5  DEN=.8667 .72198 293 END
END COMP
TRIANGPITCH  1.796 1.152 1 0  1.36 2  1.195 0  END
CASK - RODS
READ PARM RUN=YES PLT=YES NUB=YES FDN=YES NPG=1000 GEN=103 TME=60
END PARM
READ GEOM
UNIT 1
COM=!FUEL CELL!
CYLINDER   1 1 .576  173 -173
CYLINDER   0 1 .5975 173 -173
CYLINDER   2 1 .68   173 -173
UNIT 2
COM=!Not included - ROD IN CENTER!
CYLINDER   0 1 0.625 173 -173
CYLINDER   2 1 0.75 173 -173
UNIT  3
COM=!Tube + rods!
CYLINDER   0 1 4.9 173 -173
HOLE  1 0.0 1.6 0.0
HOLE  1 0.0 -1.6 0.0
HOLE  1 1.386 0.8 0.0
HOLE  1 1.386 -0.8 0.0
HOLE  1 -1.386 -0.8 0.0
HOLE  1 -1.386 0.8 0.0
HOLE  1 2.994 0.802 0.0
HOLE  1 2.994 -0.802 0.0
HOLE  1 -2.994 -0.802 0.0
HOLE  1 -2.994 0.802 0.0
HOLE  1 2.192 2.192 0.0
HOLE  1 2.192 -2.192 0.0
HOLE  1 -2.192 -2.192 0.0
HOLE  1 -2.192 2.192 0.0
HOLE  1 0.802 2.994 0.0
```

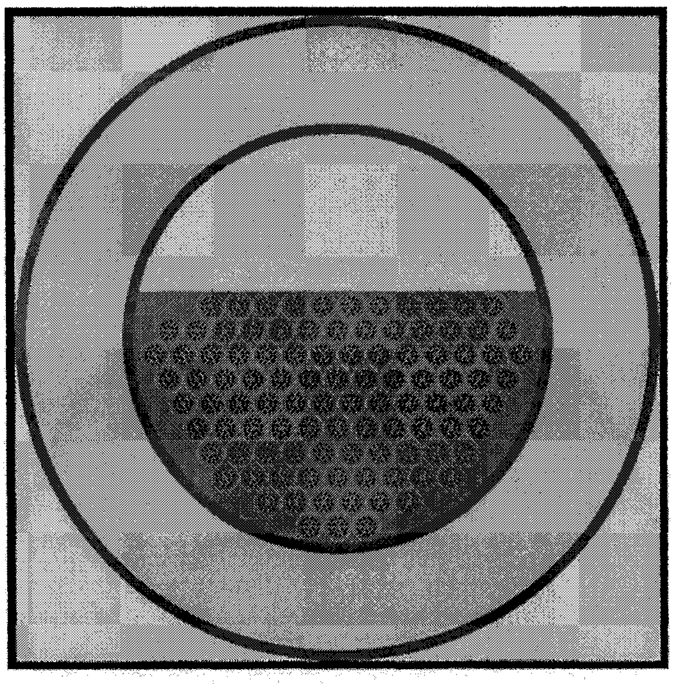


```
HOLE  1 0.802 -2.994 0.0
HOLE  1 -0.802 -2.994 0.0
HOLE  1 -0.802 2.994 0.0
CYLINDER   4 1 5.1 173 -173
GLOBAL UNIT  4
COM=!CASK!
CYLINDER   0 1 74 190.5 -190.5
HOLE  3 24.826 0.0 0.0
HOLE  3 37.239 0.0 0.0
HOLE  3 49.652 0.0 0.0
HOLE  3 62.065 0.0 0.0
HOLE  3 -24.826 0.0 0.0
HOLE  3 -37.239 0.0 0.0
HOLE  3 -49.652 0.0 0.0
HOLE  3 -62.065 0.0 0.0
HOLE  3 18.75 10.75 0.0
HOLE  3 31.25 10.75 0.0
HOLE  3 43.75 10.75 0.0
HOLE  3 56.25 10.75 0.0
HOLE  3 -18.75 10.75 0.0
HOLE  3 -31.25 10.75 0.0
HOLE  3 -43.75 10.75 0.0
HOLE  3 -56.25 10.75 0.0
HOLE  3 18.75 -10.75 0.0
HOLE  3 31.25 -10.75 0.0
HOLE  3 43.75 -10.75 0.0
HOLE  3 56.25 -10.75 0.0
HOLE  3 -18.75 -10.75 0.0
HOLE  3 -31.25 -10.75 0.0
HOLE  3 -43.75 -10.75 0.0
HOLE  3 -56.25 -10.75 0.0
HOLE  3 0.0 21.5 0.0
HOLE  3 12.413 21.5 0.0
HOLE  3 24.826 21.5 0.0
HOLE  3 37.239 21.5 0.0
HOLE  3 49.652 21.5 0.0
HOLE  3 62.065 21.5 0.0
HOLE  3 -12.413 21.5 0.0
HOLE  3 -24.826 21.5 0.0
HOLE  3 -37.239 21.5 0.0
HOLE  3 -49.652 21.5 0.0
HOLE  3 -62.065 21.5 0.0
HOLE  3 0.0 -21.5 0.0
HOLE  3 12.413 -21.5 0.0
HOLE  3 24.826 -21.5 0.0
HOLE  3 37.239 -21.5 0.0
HOLE  3 49.652 -21.5 0.0
HOLE  3 62.065 -21.5 0.0
HOLE  3 -12.413 -21.5 0.0
HOLE  3 -24.826 -21.5 0.0
HOLE  3 -37.239 -21.5 0.0
HOLE  3 -49.652 -21.5 0.0
HOLE  3 -62.065 -21.5 0.0
HOLE  3 6.25 32.25 0.0
HOLE  3 18.75 32.25 0.0
HOLE  3 31.25 32.25 0.0
HOLE  3 43.75 32.25 0.0
HOLE  3 56.25 32.25 0.0
HOLE  3 -6.25 32.25 0.0
HOLE  3 -18.75 32.25 0.0
HOLE  3 -31.25 32.25 0.0
HOLE  3 -43.75 32.25 0.0
HOLE  3 -56.25 32.25 0.0
```

```
HOLE  3 6.25 -32.25 0.0
HOLE  3 18.75 -32.25 0.0
HOLE  3 31.25 -32.25 0.0
HOLE  3 43.75 -32.25 0.0
HOLE  3 56.25 -32.25 0.0
HOLE  3 -6.25 -32.25 0.0
HOLE  3 -18.75 -32.25 0.0
HOLE  3 -31.25 -32.25 0.0
HOLE  3 -43.75 -32.25 0.0
HOLE  3 -56.25 -32.25 0.0
HOLE  3 0.0 43.0 0.0
HOLE  3 12.413 43.0 0.0
HOLE  3 24.826 43.0 0.0
HOLE  3 37.239 43.0 0.0
HOLE  3 49.652 43.0 0.0
HOLE  3 -12.413 43.0 0.0
HOLE  3 -24.826 43.0 0.0
HOLE  3 -37.239 43.0 0.0
HOLE  3 -49.652 43.0 0.0
HOLE  3 0.0 -43.0 0.0
HOLE  3 12.413 -43.0 0.0
HOLE  3 24.826 -43.0 0.0
HOLE  3 37.239 -43.0 0.0
HOLE  3 49.652 -43.0 0.0
HOLE  3 -12.413 -43.0 0.0
HOLE  3 -24.826 -43.0 0.0
HOLE  3 -37.239 -43.0 0.0
HOLE  3 -49.652 -43.0 0.0
HOLE  3 6.25 53.75 0.0
HOLE  3 18.75 53.75 0.0
HOLE  3 31.25 53.75 0.0
HOLE  3 -6.25 53.75 0.0
HOLE  3 -18.75 53.75 0.0
HOLE  3 -31.25 53.75 0.0
HOLE  3 6.25 -53.75 0.0
HOLE  3 18.75 -53.75 0.0
HOLE  3 31.25 -53.75 0.0
HOLE  3 -6.25 -53.75 0.0
HOLE  3 -18.75 -53.75 0.0
HOLE  3 -31.25 -53.75 0.0
HOLE  3 0.0 64.95 0.0
HOLE  3 12.413 64.95 0.0
HOLE  3 -12.413 64.95 0.0
HOLE  3 0.0 -64.95 0.0
HOLE  3 12.413 -64.95 0.0
HOLE  3 -12.413 -64.95 0.0
CYLINDER   6 1 78 190.5 -194.5
CYLINDER   5 1 113 190.5 -229.5
CYLINDER   6 1 117 223.5 -233.5
CUBOID     0 1     117 -117 117 -117 223.5 -233.5
END GEOM
READ BNDS ALL=REFLECT   END BNDS
READ PLOT
TTL=!RBMK CASK!
SCR=YES PIC=MIXTURE XUL=-120 YUL=120 ZUL=0 XLR=120 YLR=-120 ZLR=0
UAX=1 VAX=0 WAX=0 UDN=0 VDN=-1 WDN=0 NAX=1500 LPI=10 END
TTL=!RBMK X-Z!
SCR=YES PIC=MIXTURE XUL=-120 YUL=0 ZUL=230 XLR=120 YLR=0.56 ZLR=-240
UAX=1 VAX=0 WAX=0 UDN=0 VDN=0 WDN=-1 NAX=150 LPI=10 PLT=NO END
END PLOT
END DATA
END
```

A.6. KENOVa - Horizontal case. Tubes on bottom, void above

```
=KENOV
CASK - TUBES ON THE BOTTOM - VOID (PITCH=1.796 TRI)
READ PARM RUN=YES PLT=NO XSC=14 NUB=YES FDN=YES
          NPG=1000 GEN=103 TME=60
END PARM
READ GEOM
UNIT 1
COM=!FUEL CELL!
CYLINDER   1 1 .576  173 -173
CYLINDER   0 1 .5975 173 -173
CYLINDER   2 1 .68 173 -173
UNIT 2
COM=!ROD IN CENTER!
CYLINDER   0 1 0.625 173 -173
CYLINDER   2 1 0.75 173 -173
UNIT  3
COM=!Tube + rods!
CYLINDER   8 1 4.9 173 -173
HOLE  1 0.0 1.6 0.0
HOLE  1 0.0 -1.6 0.0
HOLE  1 1.386 0.8 0.0
HOLE  1 1.386 -0.8 0.0
HOLE  1 -1.386 -0.8 0.0
HOLE  1 -1.386 0.8 0.0
HOLE  1 2.994 0.802 0.0
HOLE  1 2.994 -0.802 0.0
HOLE  1 -2.994 -0.802 0.0
HOLE  1 -2.994 0.802 0.0
HOLE  1 2.192 2.192 0.0
HOLE  1 2.192 -2.192 0.0
HOLE  1 -2.192 -2.192 0.0
HOLE  1 -2.192 2.192 0.0
HOLE  1 0.802 2.994 0.0
HOLE  1 0.802 -2.994 0.0
HOLE  1 -0.802 -2.994 0.0
HOLE  1 -0.802 2.994 0.0
CYLINDER   4 1 5.0995 173
-173
GLOBAL UNIT  4
COM=!CASK!
ZHEMICYL-Y   8 1 74 190.5 -190.5
      CHORD  17.0
HOLE  3 0.0 -14.96 0.0
HOLE  3 10.2 -14.96 0.0
HOLE  3 20.4 -14.96 0.0
HOLE  3 30.6 -14.96 0.0
HOLE  3 40.8 -14.96 0.0
HOLE  3 51.0 -14.96 0.0
HOLE  3 61.2 -14.96 0.0
HOLE  3 -10.2 -14.96 0.0
HOLE  3 -20.4 -14.96 0.0
HOLE  3 -30.6 -14.96 0.0
HOLE  3 -40.8 -14.96 0.0
HOLE  3 -51.0 -14.96 0.0
HOLE  3 -61.2 -14.96 0.0
HOLE  3 5.1 -6.08 0.0
HOLE  3 15.3 -6.08 0.0
HOLE  3 25.5 -6.08 0.0
HOLE  3 35.7 -6.08 0.0
```

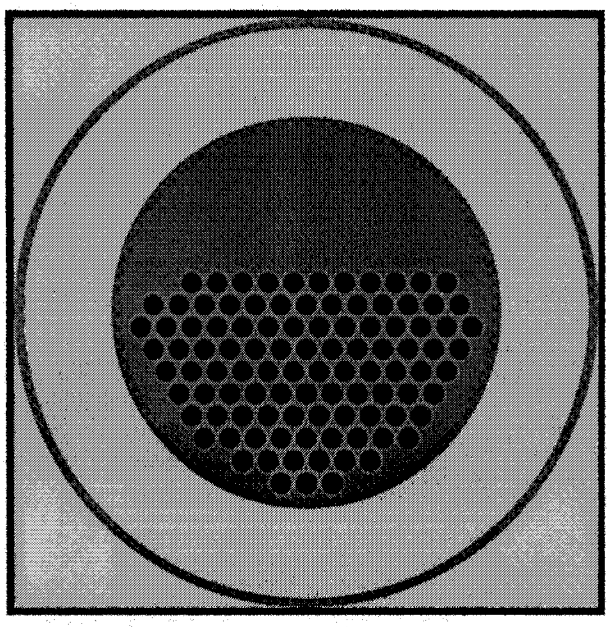


```
HOLE  3 56.1 -6.08 0.0
HOLE  3 66.3 -6.08 0.0
HOLE  3 -5.1 -6.08 0.0
HOLE  3 -15.3 -6.08 0.0
HOLE  3 -25.5 -6.08 0.0
HOLE  3 -35.7 -6.08 0.0
HOLE  3 -45.9 -6.08 0.0
HOLE  3 -56.1 -6.08 0.0
HOLE  3 -66.3 -6.08 0.0
HOLE  3 5.1 -23.8 0.0
HOLE  3 15.3 -23.8 0.0
HOLE  3 25.5 -23.8 0.0
HOLE  3 35.7 -23.8 0.0
HOLE  3 45.9 -23.8 0.0
HOLE  3 56.1 -23.8 0.0
HOLE  3 -5.1 -23.8 0.0
HOLE  3 -15.3 -23.8 0.0
HOLE  3 -25.5 -23.8 0.0
HOLE  3 -35.7 -23.8 0.0
HOLE  3 -45.9 -23.8 0.0
HOLE  3 -56.1 -23.8 0.0
HOLE  3 0.0 2.76 0.0
HOLE  3 10.2 2.76 0.0
HOLE  3 20.4 2.76 0.0
HOLE  3 30.6 2.76 0.0
HOLE  3 40.8 2.76 0.0
HOLE  3 51.0 2.76 0.0
HOLE  3 61.2 2.76 0.0
HOLE  3 -10.2 2.76 0.0
HOLE  3 -20.4 2.76 0.0
HOLE  3 -30.6 2.76 0.0
HOLE  3 -40.8 2.76 0.0
HOLE  3 -51.0 2.76 0.0
HOLE  3 -61.2 2.76 0.0
HOLE  3 0.0 -32.64 0.0
HOLE  3 10.2 -32.64 0.0
HOLE  3 20.4 -32.64 0.0
HOLE  3 30.6 -32.64 0.0
HOLE  3 40.8 -32.64 0.0
HOLE  3 51.0 -32.64 0.0
HOLE  3 -10.2 -32.64 0.0
HOLE  3 -20.4 -32.64 0.0
HOLE  3 -30.6 -32.64 0.0
HOLE  3 -40.8 -32.64 0.0
HOLE  3 -51.0 -32.64 0.0
HOLE  3 5.1 11.6 0.0
HOLE  3 15.3 11.6 0.0
HOLE  3 25.5 11.6 0.0
HOLE  3 35.7 11.6 0.0
HOLE  3 45.9 11.6 0.0
HOLE  3 56.1 11.6 0.0
HOLE  3 -5.1 11.6 0.0
HOLE  3 -15.3 11.6 0.0
HOLE  3 -25.5 11.6 0.0
HOLE  3 -35.7 11.6 0.0
HOLE  3 -45.9 11.6 0.0
HOLE  3 5.1 -41.48 0.0
HOLE  3 15.3 -41.48 0.0
HOLE  3 25.5 -41.48 0.0
HOLE  3 35.7 -41.48 0.0
HOLE  3 45.9 -41.48 0.0
HOLE  3 -5.1 -41.48 0.0
HOLE  3 -15.3 -41.48 0.0
```

```
HOLE  3 -25.5 -41.48 0.0
HOLE  3 -35.7 -41.48 0.0
HOLE  3 -45.9 -41.48 0.0
HOLE  3 0.0 -50.32 0.0
HOLE  3 10.2 -50.32 0.0
HOLE  3 20.4 -50.32 0.0
HOLE  3 30.6 -50.32 0.0
HOLE  3 40.8 -50.32 0.0
HOLE  3 -10.2 -50.32 0.0
HOLE  3 -20.4 -50.32 0.0
HOLE  3 -30.6 -50.32 0.0
HOLE  3 -40.8 -50.32 0.0
HOLE  3 5.1 -59.16 0.0
HOLE  3 15.3 -59.16 0.0
HOLE  3 25.5 -59.16 0.0
HOLE  3 -5.1 -59.16 0.0
HOLE  3 -15.3 -59.16 0.0
HOLE  3 -25.5 -59.16 0.0
HOLE  3 0.0 -68.0 0.0
HOLE  3 10.2 -68.0 0.0
HOLE  3 -10.2 -68.0 0.0
CYLINDER   0 1 74 190.5 -190.5
CYLINDER   6 1 78 190.5 -194.5
CYLINDER   5 1 113 190.5 -229.5
CYLINDER   6 1 117 223.5 -233.5
CUBOID     0 1     117 -117 117 -117 223.5 -233.5
END GEOM
READ BNDS +XB=REFLECT -XB=REFLECT +YB=REFLECT -YB=REFLECT +ZB=REFLECT
 -ZB=REFLECT  END BNDS
READ PLOT
TTL=!RBMK CASK!
SCR=YES PIC=MIXTURE XUL=-120 YUL=120 ZUL=0 XLR=120 YLR=-120 ZLR=0
UAX=1 VAX=0
WAX=0 UDN=0 VDN=-1 WDN=0 NAX=1500 LPI=10   END
TTL=!RBMK X-Z!
SCR=YES PIC=MIXTURE XUL=-120 YUL=0 ZUL=230 XLR=120 YLR=0 ZLR=-240
UAX=1 VAX=0
WAX=0 UDN=0 VDN=0 WDN=-1 NAX=700 LPI=10  END
END PLOT
END DATA
END
```

A.7. CSAS2X - Expanded, homogenised fuel, pitch 2.2 cm (maximum)

```
#CSAS2X
RBMK FUEL RODS
44GROUPNDF5                 LATTICECELL
UO2            1  DEN=10.07 1.0 293 92235 2.4 92238 97.6 END
ZIRCONIUM      2  DEN=6.44 0.99 293 END
NB             2  DEN=6.44 0.01 293 END
H2O            3  1.0 293 END
SS304          4  DEN=7.60 1.0 293 END
FE             5  DEN=1.6761 1.0 293 END
SILICON        5  DEN=0.0736 1.0 293 END
ALUMINUM       5  DEN=0.0164 1.0 293 END
CALCIUM        5  DEN=0.2532 1.0 293 END
MAGNESIUM      5  DEN=0.083 1.0 293 END
STRONTIUM      5  DEN=0.0172 1.0 293 END
BA-134         5  DEN=0.8667 .02360 293 END
CARBON         5  DEN=.0605 1.0 293 6012 100 END
SULFUR         5  DEN=.2172 1.0 293 END
OXYGEN         5  DEN=.8033 1.0 293 8016 99.96 8017 .04 END
HYDROGEN       5  DEN=.0223 1.0 293 END
CARBONSTEEL    6  DEN=7.60 1.0 293 END
IRON           7  DEN=7.60 1.0 293 END
H2O            8  1.0 293 END
BA-135         5  DEN=0.8667 0.06477 293 END
BA-136         5  DEN=0.8667 0.07769 293 END
BA-137         5  DEN=0.8667 .11196 293 END
BA-138         5  DEN=.8667 .72198 293 END
END COMP
TRIANGPITCH  2.200 1.152 1  3  1.36 2  1.195 0  END
CASK - HOMOGENISED RODS (PITCH=2.200 TRI)
READ PARM RUN=YES PLT=YES NUB=YES FDN=YES
          NPG=1000 GEN=103  TME=60
END PARM
READ GEOM
UNIT 1
COM=!FUEL CELL!
CYLINDER   1 1 .576 173 -173
CYLINDER   0 1 .5975 173 -173
CYLINDER   2 1 .68 173 -173
UNIT 2
COM=!Not used - ROD IN CENTER!
CYLINDER   0 1 0.625 173 -173
CYLINDER   2 1 0.75 173 -173
UNIT  3
COM=!Tube + rods!
CYLINDER  500 1 4.9 173 -173
CYLINDER    4 1 5.0995 173 -173
GLOBAL UNIT  4
COM=!CASK!
CYLINDER   8 1 74 190.5 -190.5
HOLE  3 0.0 -14.96 0.0
HOLE  3 10.2 -14.96 0.0
HOLE  3 20.4 -14.96 0.0
HOLE  3 30.6 -14.96 0.0
HOLE  3 40.8 -14.96 0.0
HOLE  3 51.0 -14.96 0.0
HOLE  3 61.2 -14.96 0.0
HOLE  3 -10.2 -14.96 0.0
HOLE  3 -20.4 -14.96 0.0
HOLE  3 -30.6 -14.96 0.0
```



```
HOLE  3 -40.8 -14.96 0.0
HOLE  3 -51.0 -14.96 0.0
HOLE  3 -61.2 -14.96 0.0
HOLE  3 5.1 -6.08 0.0
HOLE  3 15.3 -6.08 0.0
HOLE  3 25.5 -6.08 0.0
HOLE  3 35.7 -6.08 0.0
HOLE  3 45.9 -6.08 0.0
HOLE  3 56.1 -6.08 0.0
HOLE  3 66.3 -6.08 0.0
HOLE  3 -5.1 -6.08 0.0
HOLE  3 -15.3 -6.08 0.0
HOLE  3 -25.5 -6.08 0.0
HOLE  3 -35.7 -6.08 0.0
HOLE  3 -45.9 -6.08 0.0
HOLE  3 -56.1 -6.08 0.0
HOLE  3 -66.3 -6.08 0.0
HOLE  3 5.1 -23.8 0.0
HOLE  3 15.3 -23.8 0.0
HOLE  3 25.5 -23.8 0.0
HOLE  3 35.7 -23.8 0.0
HOLE  3 45.9 -23.8 0.0
HOLE  3 56.1 -23.8 0.0
HOLE  3 -5.1 -23.8 0.0
HOLE  3 -15.3 -23.8 0.0
HOLE  3 -25.5 -23.8 0.0
HOLE  3 -35.7 -23.8 0.0
HOLE  3 -45.9 -23.8 0.0
HOLE  3 -56.1 -23.8 0.0
HOLE  3 0.0 2.76 0.0
HOLE  3 10.2 2.76 0.0
HOLE  3 20.4 2.76 0.0
HOLE  3 30.6 2.76 0.0
HOLE  3 40.8 2.76 0.0
HOLE  3 51.0 2.76 0.0
HOLE  3 61.2 2.76 0.0
HOLE  3 -10.2 2.76 0.0
HOLE  3 -20.4 2.76 0.0
HOLE  3 -30.6 2.76 0.0
HOLE  3 -40.8 2.76 0.0
HOLE  3 -51.0 2.76 0.0
HOLE  3 -61.2 2.76 0.0
HOLE  3 0.0 -32.64 0.0
HOLE  3 10.2 -32.64 0.0
HOLE  3 20.4 -32.64 0.0
HOLE  3 30.6 -32.64 0.0
HOLE  3 40.8 -32.64 0.0
HOLE  3 51.0 -32.64 0.0
HOLE  3 -10.2 -32.64 0.0
HOLE  3 -20.4 -32.64 0.0
HOLE  3 -30.6 -32.64 0.0
HOLE  3 -40.8 -32.64 0.0
HOLE  3 -51.0 -32.64 0.0
HOLE  3 5.1 11.6 0.0
HOLE  3 15.3 11.6 0.0
HOLE  3 25.5 11.6 0.0
HOLE  3 35.7 11.6 0.0
HOLE  3 45.9 11.6 0.0
HOLE  3 56.1 11.6 0.0
HOLE  3 -5.1 11.6 0.0
HOLE  3 -15.3 11.6 0.0
HOLE  3 -25.5 11.6 0.0
HOLE  3 -35.7 11.6 0.0
```

```
HOLE  3 -45.9 11.6 0.0
HOLE  3 5.1 -41.48 0.0
HOLE  3 15.3 -41.48 0.0
HOLE  3 25.5 -41.48 0.0
HOLE  3 35.7 -41.48 0.0
HOLE  3 45.9 -41.48 0.0
HOLE  3 -5.1 -41.48 0.0
HOLE  3 -15.3 -41.48 0.0
HOLE  3 -25.5 -41.48 0.0
HOLE  3 -35.7 -41.48 0.0
HOLE  3 -45.9 -41.48 0.0
HOLE  3 0.0 -50.32 0.0
HOLE  3 10.2 -50.32 0.0
HOLE  3 20.4 -50.32 0.0
HOLE  3 30.6 -50.32 0.0
HOLE  3 40.8 -50.32 0.0
HOLE  3 -10.2 -50.32 0.0
HOLE  3 -20.4 -50.32 0.0
HOLE  3 -30.6 -50.32 0.0
HOLE  3 -40.8 -50.32 0.0
HOLE  3 5.1 -59.16 0.0
HOLE  3 15.3 -59.16 0.0
HOLE  3 25.5 -59.16 0.0
HOLE  3 -5.1 -59.16 0.0
HOLE  3 -15.3 -59.16 0.0
HOLE  3 -25.5 -59.16 0.0
HOLE  3 0.0 -68.0 0.0
HOLE  3 10.2 -68.0 0.0
HOLE  3 -10.2 -68.0 0.0
CYLINDER   6 1 78 190.5 -194.5
CYLINDER   5 1 113 190.5 -229.5
CYLINDER   6 1 117 223.5 -233.5
CUBOID     0 1     117 -117 117 -117 223.5 -233.5
END GEOM
READ BNDS +XB=REFLECT -XB=REFLECT +YB=REFLECT -YB=REFLECT +ZB=REFLECT
 -ZB=REFLECT  END BNDS
READ PLOT
TTL=!RBMK CASK!
SCR=YES PIC=MIXTURE XUL=-120 YUL=120 ZUL=0 XLR=120 YLR=-120 ZLR=0
UAX=1 VAX=0
WAX=0 UDN=0 VDN=-1 WDN=0 NAX=1000 LPI=10 PLT=YES END
END PLOT
END DATA
END
```