October 22, 2002

MEMORANDUM TO: Jack Rosenthal, Chief, SMSAB:DSARE:RES

FROM: Frank Odar, SMSAB:DSARE:SMSAB **Original signed by F. Odar**

SUBJECT: REPORT ON IDENTIFICATION OF COMPUTER CODE NEEDS FOR GAS COOLED REACTORS

Attached is the report entitled, “Identification of Thermal Hydraulic Code Needs for Analysis of Transients and Accident in Gas Cooled Reactors.” This report has been prepared after the review of earlier regulatory issues on the integrity of TRISO coated particles and identifying areas where knowledge lacked and conservatisms were imposed and present status of research efforts in different countries.

The report recommends the development of two computer code modules: 1) Fuel Element Performance Code/Module, and 2) Dust&Transport&Deposition&Liftoff Code Module in addition to upgrades to the TRAC-M code. It is recommended that development of these code modules be included in long range planning and detailed requirements for these codes/modules be established.

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# IDENTIFICATION OF THERMAL HYDRAULIC CODE NEEDS FOR ANALYSIS OF TRANSIENTS AND ACCIDENTS IN GAS COOLED REACTORS

**Frank Odar**

**September 2002**

## EXECUTIVE SUMMARY


Reviews of some regulatory issues concerning TRISO particles and status of knowledge based on recent operating experience were performed. These reviews showed that there are issues and areas where uncertainties exist and where research efforts should be directed. Regulatory issues where uncertainties exist, have been resolved in a conservative way. For example, uncertainties concerning integrity of TRISO particles have contributed to the selection of a low leakage type of containment for the Ft. StVrain plant.

Since then, substantial amount of research on the integrity of TRISO particles has been performed. Hence, the uncertainties can be reduced. In the PIRT program of TRISO particles which is underway, the extent of knowledge and the areas where additional research is needed, will be determined.

In terms of thermal hydraulic codes, TRAC-M is suitable for modifications to predict transients and accidents in gas cooled reactors. Limited amount of modifications will be required to predict the phenomena in gas cooled reactors. However, major efforts are required to develop a fuel element module for prediction of fission product release from TRISO particles. Development of a second major module for prediction of dust transport, its deposition and liftoff and fission product plateout and washoff is also necessary.

It is necessary to develop detailed requirements documents for these modules after the PIRT program on TRISO particles is completed. Detailed requirements documents accurately define the scope of work and permit estimation of resources.

## REGULATORY PERSPECTIVES

Ft. StVrain plant has been licensed, operated and now it is shutdown. The staff had to use conservative criteria because of lack of knowledge on the integrity of TRISO coated particles. This lack of knowledge contributed to selection of a low leakage type of containment. Now, there is a new interest among vendors to construct and operate gas cooled reactors. A preapplication may be submitted by General Atomic to build a gas cooled reactor.

This report identifies areas of uncertainties in integrity of TRISO coated particles and recommends development of two computer code modules, **“Fuel Element Performance Code / Module”** and **“Dust Transport & Deposition & Liftoff Code Module”** and modification of the **TRAC-M** code. Consequences of success or failure of developing these codes will lead to whether or not the NRC staff will have capability of auditing vendor submittals to make sound technical judgements on licensing issues. Development of these computer code modules attempts to provide clear and sufficient knowledge to either confirm or deny analyses put forward by the industry.

Development of these codes will improve on the agency’s performance relative to the Performance Goal. Development of these codes based on new test data will reduce conservatism. Reducing conservatism while keeping plant operations safe, reduces unnecessary regulatory burden and improves effectiveness and efficiency. The consequences of not having these codes developed are increased conservatism, cost and regulatory burden and reduced efficiency.

# 1. INTRODUCTION

## 1.1 Identification of Needs:

The first step in a code development process is to identify the phenomena which are to be modeled by the code. This is usually done by a PIRT process. PIRT stands for "Phenomena Identification and Rating Table". During the PIRT process, first the plant type is selected. This provides information on the design of systems and operational characteristics of the plant. This information is used to predict the phenomena occurring during a postulated transient or accident. Analysis of risk-dominating events determines transients and accidents to be considered. The PIRT process identifies the phenomena and assigns rating of importance for each phenomena. These importance rating factors are used to in many ways. They are used to 1) identify research needs, 2) prioritize code development and experimental programs and 3) prioritize code validation efforts.

At this point in time, there is no specific application for licensing of a gas cooled reactor. Hence, the plant type is not known and the design characteristics including systems and operational characteristics are not known. The phenomena and issues affecting the safe operation of these reactors are expected to be different for each type of plant. Because there is no established design, the identification of needs, at this point in time, cannot address all of the details of phenomena which may occur in a specific design. Rather, one discusses the general nature of the phenomena that may occur in these reactors and suggests technical approaches for solution of problems. It is suggested that once the plant type is known, the PIRT process should be performed to confirm the general nature of phenomena addressed here and also identify other specific phenomena which may be associated with a specific reactor type. At this point in time, two classes of plants are considered. They are Pebble Bed Modular Reactors (PBMRs) and prismatic type high temperature reactors (HTRs).

## 1.2 TRISO Coated Fuel Particles

Regardless of which type of reactor is submitted for licensing, NRC will be interested in quantifying the fission product release during normal operations and accidents. Modern gas cooled reactors utilize TRISO coated fuel particles. The particle is composed of several layers of materials. Fig.1 depicts a TRISO coated fuel particle. The kernel contains the fissionable

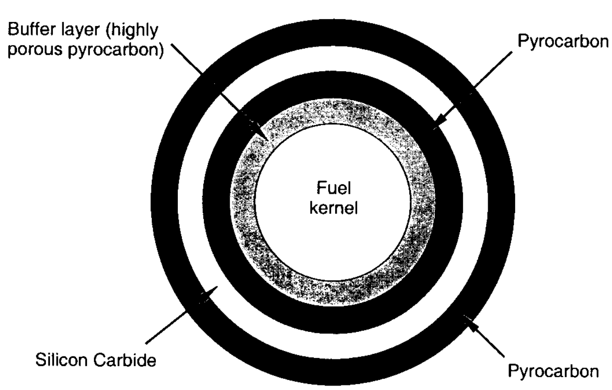


Figure 1 - TRISO Coated Fuel Particle

material.  As the fission process takes place, the power and fission products are produced.  The buffered carbon layer provides volume for gaseous fission products.  Silicone Carbide layer contains these gaseous fission products; i. e., it almost serves as a containment.

The functions of these various layers can be summarized as follows: The fuel kernel provides fission energy and retains short lived fission products.  The buffer layer (porous carbon layer) attenuates fission recoils and provides volume for gaseous fission products.  The inner pyrocarbon layer provides substrate for silicone carbide during the manufacturing process.  The silicone carbide layer retains gas and metal fission products.  The outer pyrocarbon layer provides a bounding surface to graphite matrix and provides some fission product barrier in particles with defective silicone carbide layer.  Some of the fission products are absorbed in the graphite matrix.

It is clear that the fission product release is highly dependent on the integrity of the TRISO coated fuel particles particularly that of the silicone carbide layer.  Manufacturing defects in the silicone carbide layer are one class of the major causes of fission product release during normal operations.  The manufacturing process does not provide 100% integrity for the TRISO coated particles.

The fuel element in a PBMR is a spherical pebble.  It contains about 15,000 TRISO particles.  Each particle consists of a spherical $UO_2$ kernel coated by concentric layers of material discussed above.  The size of a TRISO particle is about 0.9mm in diameter.  All of these particles are uniformly distributed in a graphite matrix material which forms the inner-fuel containing zone of 50mm diameter.  This zone is surrounded by a 5mm thick  graphite material.  In case of prismatic high temperature reactors, TRISO particles are formed into fuel compacts and inserted into graphite fuel elements.

As the power is produced in TRISO particles, the heat is transferred via conduction inside the fuel element and conducted, convected and radiated from the surfaces.  During normal operations heat is carried away to operate the gas turbine.   During accidents where the primary flow stops, heat is conducted and radiated to the edge of the core, conducted across the graphite reflectors and the vessel and convected and radiated via Reactor Cavity Cooling System.

## 2.0 REVIEW OF SOME REGULATORY ISSUES

### 2.1 Barriers to Fission Product Release

There are five barriers reducing the fission product release to the environment. They are:

1. Fuel kernels of the TRISO coated fuel particles,
2. TRISO particle coatings,
3. Matrix graphite,
4. Primary system boundary,
5. Containment (low leakage or vented)

Effectiveness of these barriers depends on the chemistry and half lives of the various radionuclides, irradiation effects and type of events. Clearly, if the containment is a vented type, it will not provide as good a barrier as the low leakage type. If a primary system break is postulated, the other barriers for the fission product release are the first three barriers which provide integrity for the fuel elements. This may put stringent requirements on the manufacture and performance of TRISO coated fuel particles to ensure their integrity.

All phenomena related to fission product release and retention in these barriers should be completely understood and related calculations be validated. The fission product retaining phenomena, if not completely understood and validated, may be neglected entirely to ensure conservatism of fission product release. As discussed below, some of these phenomena are not well understood, Ref. 1. Examples are 1) fission product sorption from the gas phase by graphitic core components and 2) fission product retention in the confinement if it is vented. Figure 2 adapted from Ref. 2 illustrates the concept of these various barriers.

### 2.2 Fission Product Release from TRISO Particles

This section will review the effectiveness TRISO coating in providing a barrier for fission product release. Most of the material is presented in Ref. 3 in detail. In Ref. 3 the staff has reviewed General Atomic, GA, submittals and presented their findings. It is useful to present a short review of these findings since these findings show where the uncertainties are.

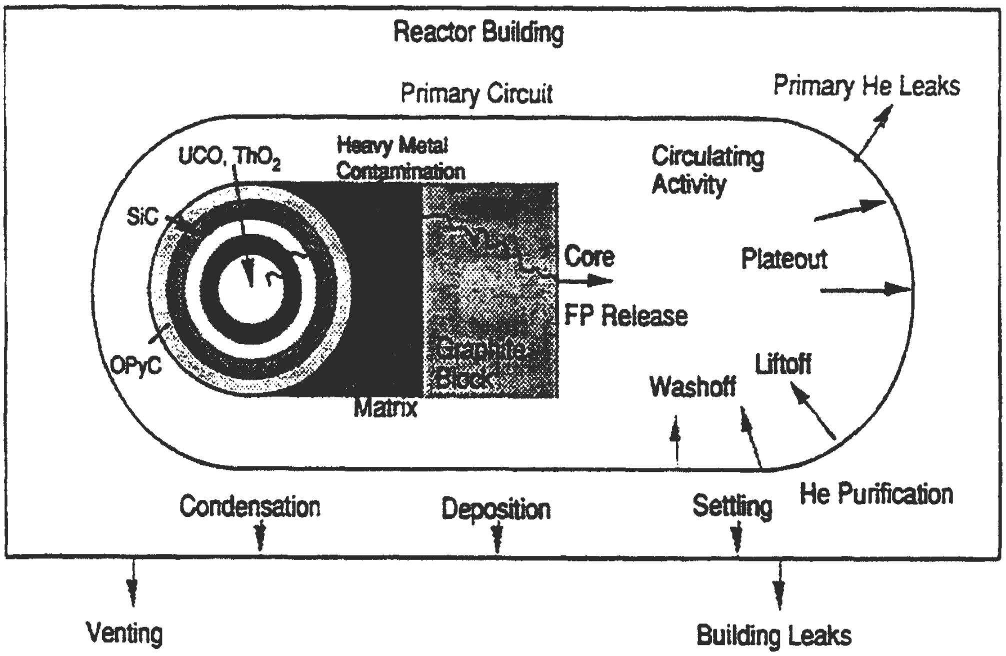


Figure 2 - Fission Product Barriers

The GA model on fission product release mechanisms had four main parts: 1) fuel particle coating failure, 2) release from fuel particles, 3) transport through graphite, and 4) release to coolant. These mechanisms are illustrated in Fig. 3. As shown in the figure, although the coating may be intact, some release will still occur. Fig. 4 illustrates the release of various types of fission products with and without coating failure. The release of fission products with

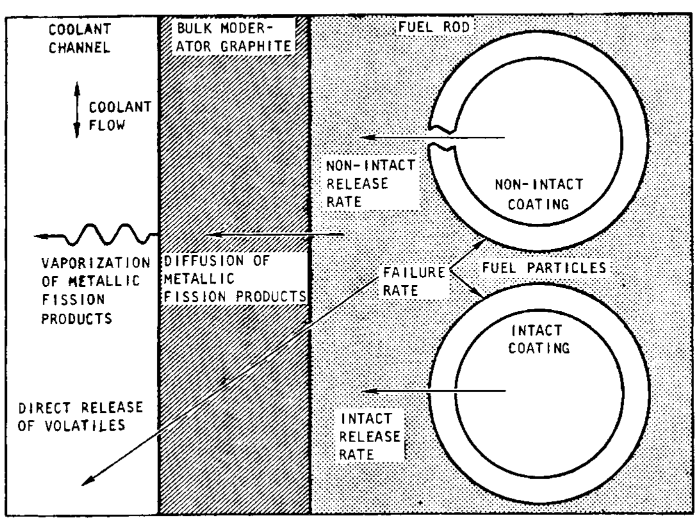


Figure 3 - Fission Product Release Mechanisms from Fuel Particle to Primary Coolant (From Ref. 3)

coating failure is several orders of magnitude higher than  with the intact coating.

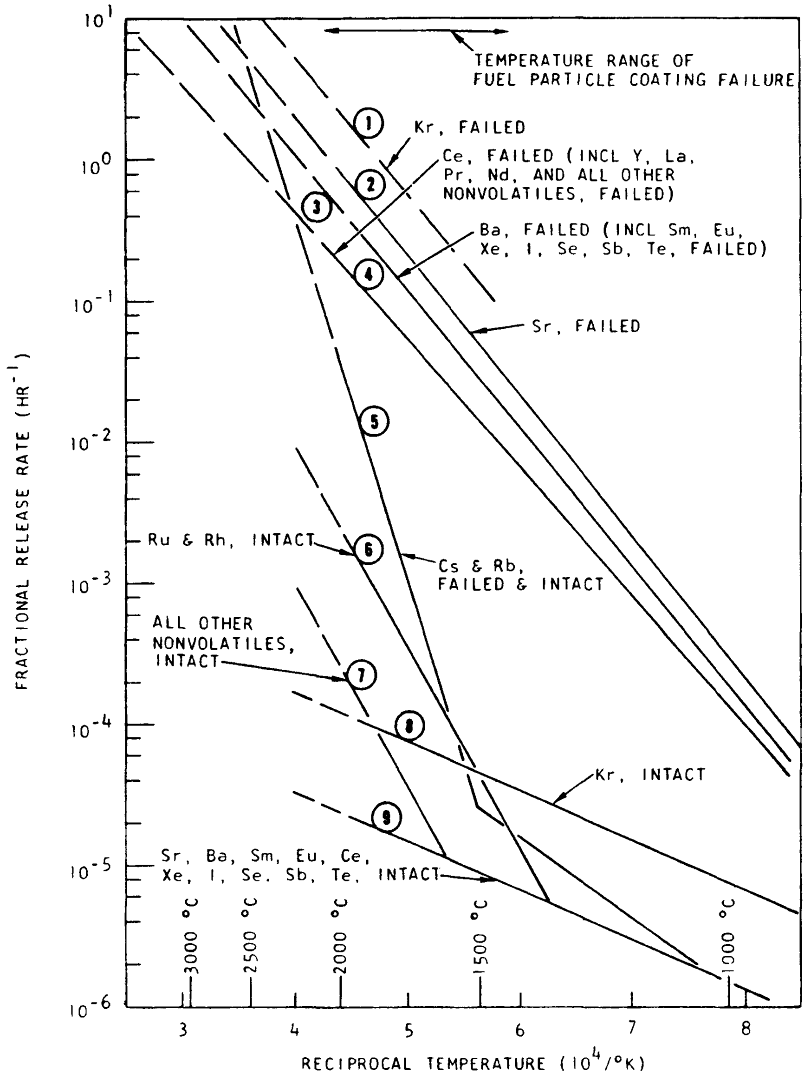


Figure 4 - Fission Product Release from Intact and Failed Particles (Ref. 3)

The fuel particle coating failure mechanism is the most complex portion of the overall release model.  The coating failure depends on the temperature of the fuel attained during the accident and the irradiation time.  Figure 5 presents the relationship of coating failures with fuel temperature and irradiation time.

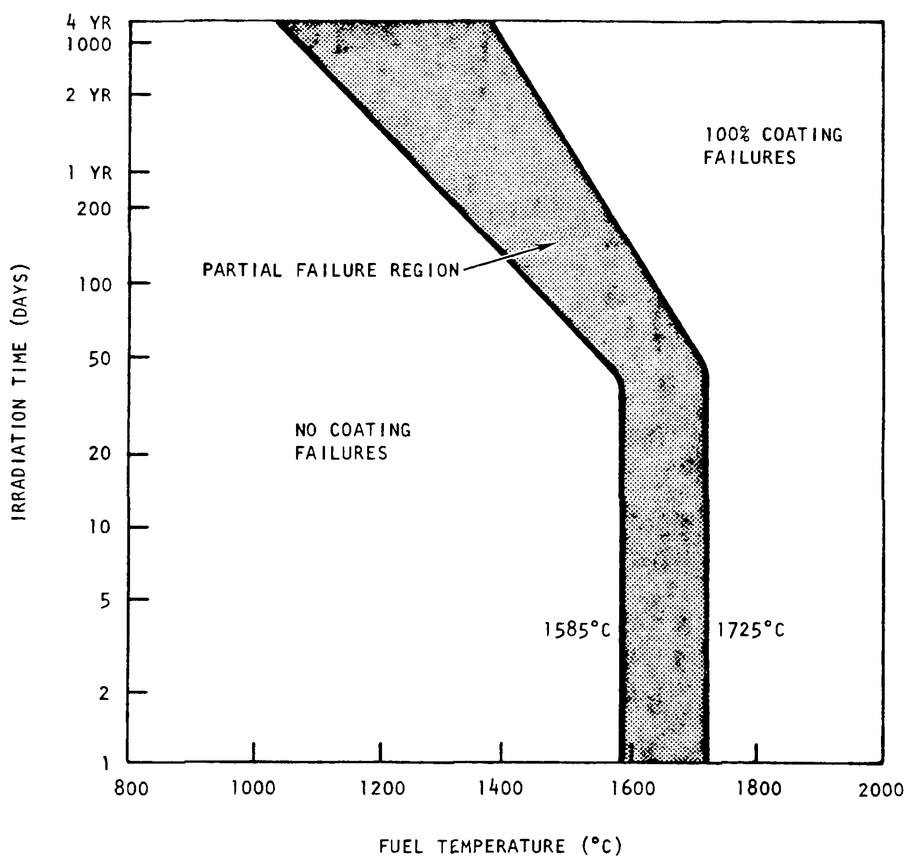


Figure 5 - TRISO Fuel Particle Coating Diagram, (Ref. 3)

These fuel temperatures were coupled with calculations of the temperature history of the hottest 1/4 of the core during a loss-of-forced circulation (LOFC) accident to provide an “interim” model for fission product release.  Further, as a simplifying approximation the time dependent release was assumed to occur in two distinct steps:

1. The first step of 40% of the total release at one hour after shutdown and loss-of-cooling, and

2. A second step of 60% of the total release at two hours after shutdown

The “interim” model assumed that the particles failed when the temperature reached the lower boundary of the failure curve in Fig. 5. This was very conservative. Additional data were obtained. The new data provided some understanding of failure mechanisms in the “Partial Failure Region”.

The analytical model treats the coated particles as composite pressure vessels in which internal pressure is built up during irradiation by the accumulation of gaseous fission products in the void volume of the buffer layer. As the pressure inside the silicone carbide layer increases and the layer expands, the outside pyrocarbon layer provides some compression to the silicone carbide layer since it does not expand as much. To a first approximation, the fuel designers assumed that uncertainties in the fuel kernel diameter and in thicknesses of various layers control the internal pressure; and consequently, induced stresses. Figure 7 shows distribution of fuel kernel diameters and buffer coating thicknesses for a “typical” batch of coated particles. Figure 6 illustrates results of GA calculations of stresses in silicone carbide layer using the same distributions shown in Figure 7. These calculations were performed for end-of-life conditions where irradiation effects; i. e., fission product release, was maximized.

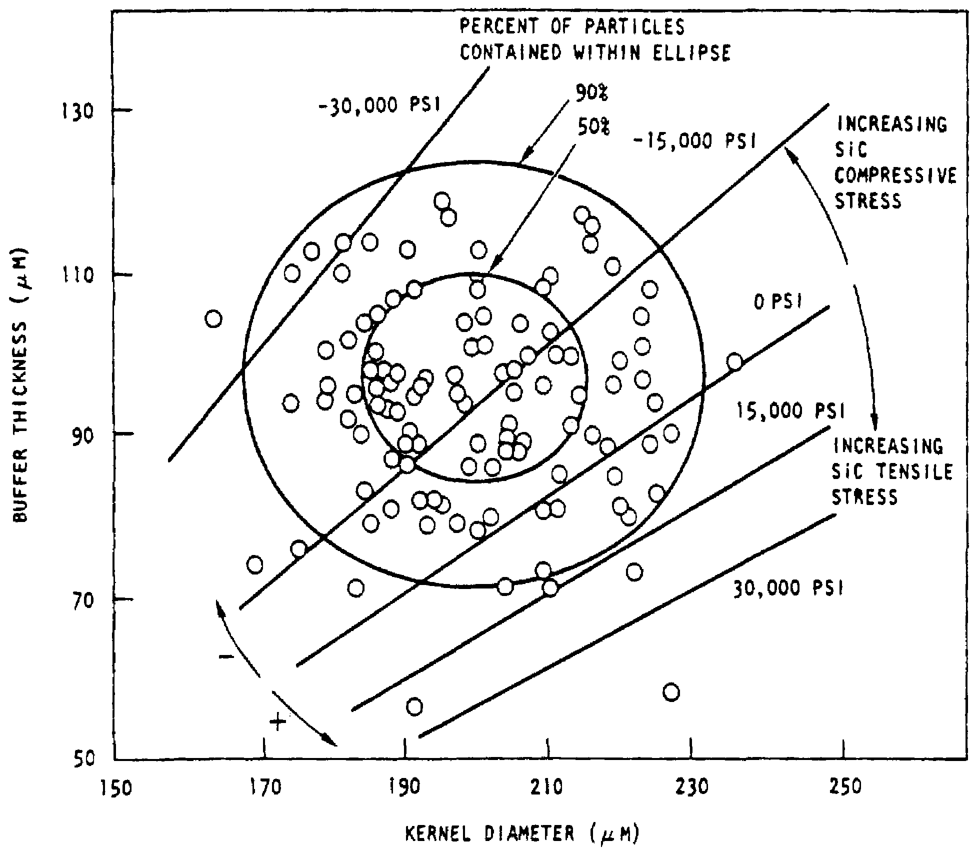


Figure 6 - Stresses in the Silicone Carbide Layer Given the Distribution of Kernel Diameters and Buffer Thicknesses for the TRISO Coated Fuel Particle Batch

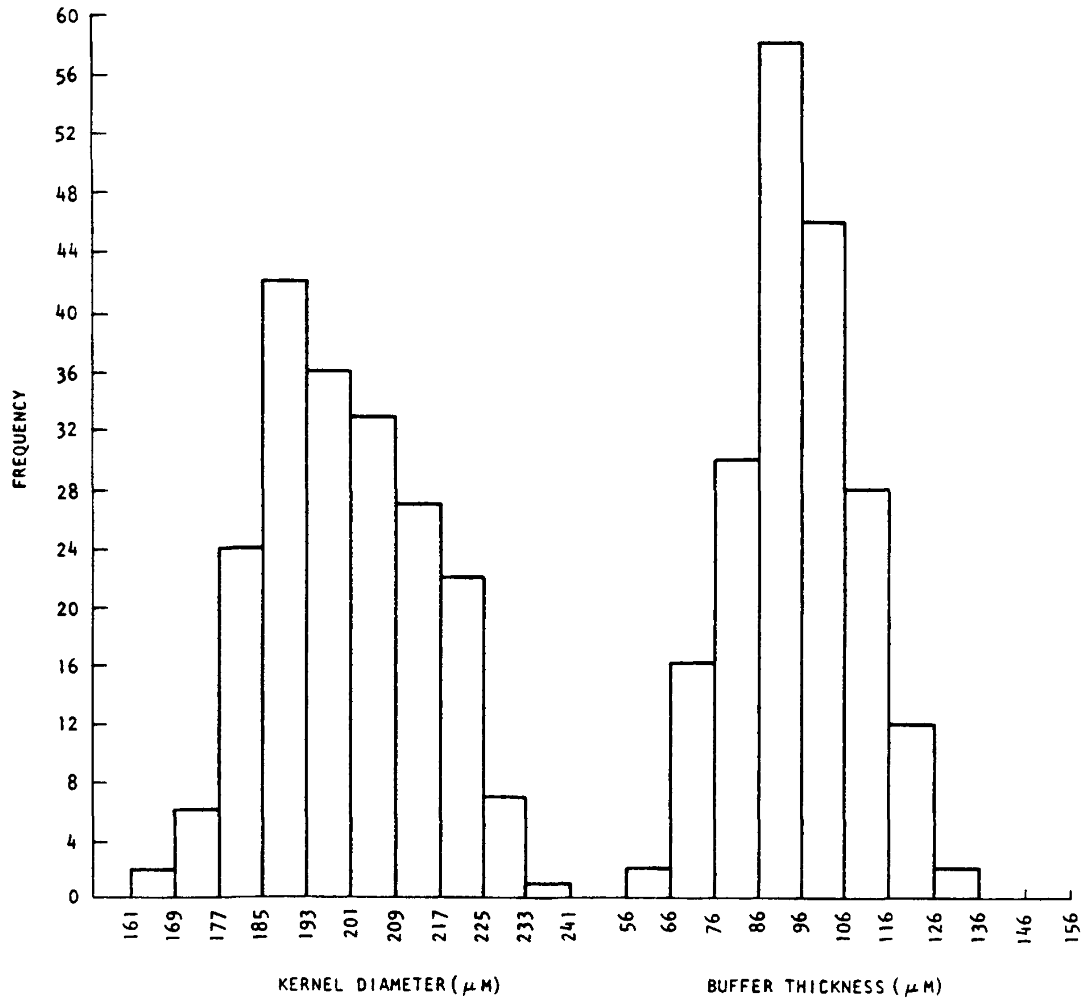


Figure 7 - Distribution of Fuel Kernel Diameters and Buffer Coating Thicknesses for a Typical TRISO Particle Batch

Other than fabrication defects, the important fuel failure mechanism is that so-called "pressure vessel failure." This occurs when the stresses induced by fission gas pressure within the particles exceed the strength of the particle coatings. The coating may be weakened because of the effect of certain phenomena that occur during irradiation. Some of these phenomena affecting fission product release from TRISO coated particles are listed below:

1. <u>Amoeba Effect (Kernel Migration)</u>

Unidirectional migration of the fuel kernel through the coating up the temperature gradient may occur. The kernel can migrate through the buffer layer and may interact and penetrate the outer coating layers resulting in coating failure and fission product release.

2. <u>Pressure Vessel Failure</u>

The effects of fuel temperature, neutron exposures, kernel burnup, kernel and coating dimensions and densities are considered in calculating pyrocarbon (PyC) and silicone carbide (SiC) layer stresses. As discussed above, irradiation-induced stresses in the TRISO coated particles depend on kernel diameter and buffer thickness. Given these two parameters, and the temperature, GA estimated the vessel failure fraction as a function of fast neutron fluence and burnup as shown in Figure 8. Since the relationship between fast neutron exposure and kernel burnup will vary in an operating HTGR from point to point in the core as the fast-to-thermal neutron flux varies, GA assumed the larger of two failure fractions to describe the fuel behavior.

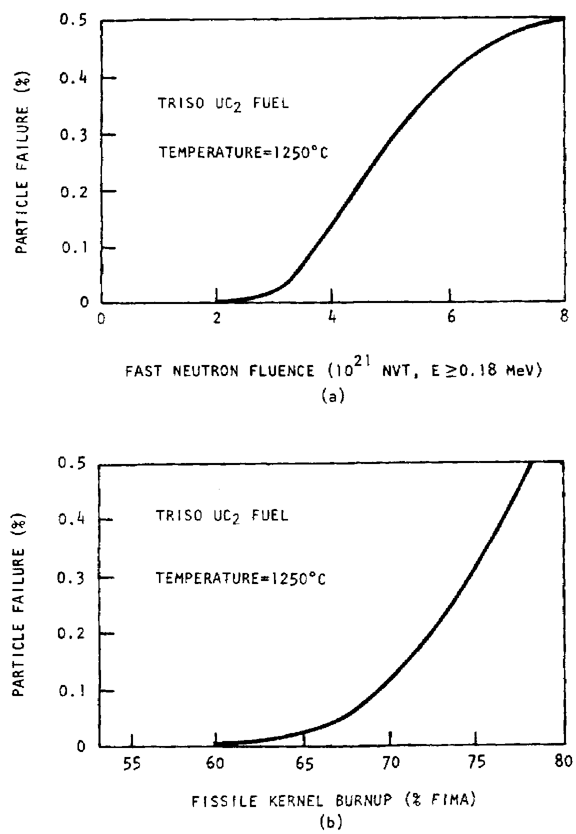


Figure 8 - Assumed Pressure Vessel Failure Fraction vs a) Fast Neutron Fluence and b) Kernel Burnup for TRISO Fuel

The selection of 1250 $^{0}$C is important since this temperature is a boundary for selection of different assumptions and models. When temperature is below this value different assumptions and models would apply. According to GA, as a result of the work depicted in Figure 8, TRISO coated fuel particle failures would be less than 1% of the fissile fuel. This would also indicate that SiC related stresses are more than 30,000 psi for less than 1% of the fissile fuel at temperatures of 1250 $^{0}$ C . GA argued that since quality control plan used to evaluate HTGR fuel was based on 95% confidence level, small fractions of the fuel would not meet the specification. GA proposed that 50% of the fuel particles having calculated SiC stresses exceeding 30,000 psi fail during operation at 1250 $^0$C.

The NRC staff did not accept the proposal because of the non-prototypicality of the sample that GA used, uncertainties in fabrication methods of the particles and the possible effects of irradiation time. The irradiation creep and relaxation of the outer pyrocarbon under stress could relieve the compressive force on the SiC layer, causing the SiC to be subjected to a higher differential pressure from the gas inside the particle. In addition, the staff suggested that the visual method of failure detection which was used by GA, was generally not accurate since it can miss small flaws. SiC is a brittle material which has a distribution of fracture strengths determined by the flaw size distribution rather than a unique strength value. In addition to the 30,000psi SiC strength limit; i. e., the particles with calculated stress level of 30,000 psi would fail, the staff proposed that particles should be fabricated using a quality control program which would provide coating dimensions assuring that SiC layer for a minimum of 80% of the TRISO particles would always be under compression throughout the life of the core under normal operating conditions.

For fuel temperatures above 1250 $^0$C, there are more uncertainties in describing pyrocarbon dimensional changes and creep. At around 1500 to 1600 $^0$C another phenomena, SiC - fission product attack becomes observable. For the range of 1250 $^0$C to 1600 $^0$C, GA proposed to estimate the effect of a rapid temperature transient by superimposing the effects of an instantaneous temperature increase on SiC layer stress calculations at any neutron exposure. The data in this region were very sparse. The NRC staff agreed with this approach provided that the failure of 100% of fuel particles having calculated SiC layer stresses exceeding 30,000 psi was assumed and that failure rate was less than 0.25% per year.

3. Failure Induced by Fission Product Attack of the SiC Layer

At temperatures above ~1500 t0 1600 $^0$C, SiC-fission product chemical interactions weaken the SiC layer, thereby rendering it susceptible to failure at stresses below 30,000 psi. The failure of the particles increases with increases in temperature and burnup. Predictions of TRISO coated particle failures cannot be made as was done at lower temperatures where the temperatures could be calculated using thermal hydraulic codes and SiC stresses using these calculated temperatures and irradiation. This is because it is very difficult, if not impossible, to determine the extent of weakening of the SiC layer.

Since the data were very sparse, a bounding type conservative correlation was developed by GA. This correlation is illustrated in Figure 9.

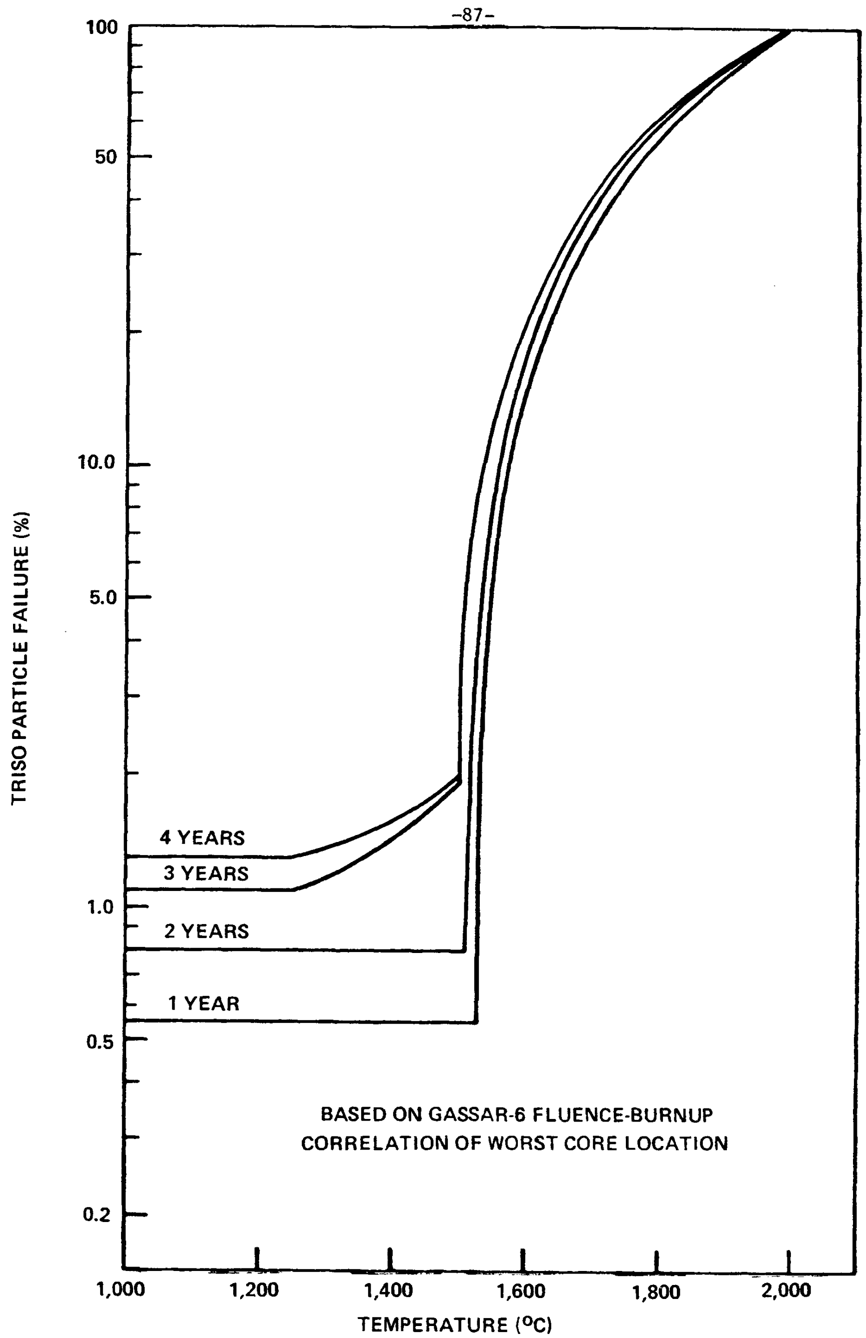


Figure 9 - TRISO Particle Coating Failure

Fig. 9 assumes that all refueling regions are at the end of cycle in order to maximize burnup and that all locations in the core experience the fluence and burnup of the worse core location; i. e., those locations of 1, 2, 3, and 4 year old fuel representing less than 1% of the core. This figure depicts the fuel failure model containing all conservative requirements and assumptions discussed above. The horizontal portions of the curve for each fuel age were obtained from the sums of fuel fractions for defective fuel and fuel operating under temperature of 1250 $^0$C. The defective fuel fraction was assumed to be 0.3% at BOL. Since a minimum failed fuel fraction for operations under 1250 $^0$C was taken to be 0.25% per year, failed fuel fractions are 0.55%, 0.80%, 1.05% and 1.3% for 1, 2, 3, and 4 year fuel respectively.

### 2.3 Normal Reactor Operations (Fuel Temperatures less than 1250 $^0$C)

As discussed above, some fission products are released from the core into the primary system because of the defective TRISO particles and also because of particle failures when tensile stresses in the SiC layer exceed 30,000 psi. There may also be some failures because of amoeba effect. These fission products are in form of gaseous and metallic fission products . As shown in Figures 2 and 3, they diffuse through the graphite matrix and the graphite which forms the outer layer of the fuel element (in case of pebbles) or the fuel compact (in case of prisms). Some gaseous fission products go through the graphite matrix and the graphite layer and enter into the primary system. Most of the metallic fission products are retained in the graphite but some of them go through. They may vaporize at the surface of the fuel element. Both the vaporized metallic and gaseous fission products are carried away by helium coolant. The "free" activities from these products are in atomic or molecular form. The helium purification system removes some of these fission products (noble fission gases) from the primary coolant at a rate determined by the gas flow through the purification system. Some of the vaporized metallic fission products may condense on cooler surfaces in the primary system. Deposition of these fission products at various surfaces is called the "plate-out" process which is shown in Figure 2 adapted from Ref. 2..

Another important issue related to fission product release during normal operations is dust production, its transportation and deposition. In PBMRs dust is produced because of abrasion of fuel elements as they move downward in the core by touching each other and rubbing side reflector walls and also during recirculation of the fuel elements in the fuel handling system. It is expected that dust formation in HTGRs with prismatic type fuel elements (compacts) will be less depending on the design of the system. The vapor phase metallic fission products released to helium stream, as explained above, may condense and bound to dust. Since the dust is a product of abrasion from fuel elements, some metallic fission products already reside in dust particles. The dust which is transported through the primary system, may adhere on some primary system surfaces and may settle in quiescent coolant regions. The dust which settles in quiescent regions or adheres surfaces over a long period of time, will form layers of sediment containing different fission product materials. These layers exhibit a gradient of activities because of not only radioactive decay of a specific material but also existence of different radioactive decay characteristics of different materials.

The third issue in fission product release during normal operation is understanding the interaction of dust with molecular gas-borne or plated-out activity. According to experiments in AVR, dust from quiescent coolant regions obtained during blower transients has about the same specific activity as adhesively bound dust; however, these activities are much higher than found on fuel element surfaces, Ref. 2.

In summary, there are large uncertainties in quantifying the amount of dust production, its transport and deposition and in understanding of interaction of dust with molecular gas-borne or plate-out activity in PMBRs. Some of the dust can be successfully removed via filtering system. The amount of removal depends on the design and operation of the filtering system. However, 100% removal should not be expected. A lot more research in quantifying the dust production, its transport, deposition and interaction with gas-borne and plated-out activities is necessary, Refs. 1 and 2.

## 2.4 Transient and Accidents

During an accident where temperatures of the TRISO coated particles increase, there are more failures of the particles. The failure rate depends on the temperature and the neutron fluence as shown in Fig. 9. Although the basis of Figure 9 is very conservative, similar figures for other reactors can be established on a best estimate basis provided there are enough data and knowledge to remove conservatisms.

Fig. 9 is the type of a figure necessary for calculation of the fission product release induced by a heat-up transient. The thermal-hydraulic codes would calculate the average temperature of TRISO coated fuel particles. Preferably average fuel temperatures of separate regions of the core will be calculated using these codes. This will provide information on temperature gradients which can be used to calculate amoeba effect. Using these temperatures and the burnup information and a figure, similar to Fig. 9, established from a test data from a specific manufacturing process, one can calculate percentage of TRISO coated particle failures for a specific transient and specific reactor core design. Once percentages of fuel particle failures as a function of fuel temperature and fluence are known, the source term including its specific composition, activity, and form can be calculated.

During transients and accidents particularly where primary coolant loss occur, the “liftoff” phenomena should be considered. The condensible nuclides which are plated-out in the primary system, may be partially reentrained and released into the reactor building or containment during rapid depressurization. This mechanical reentrainment of deposited particulate matter contaminated by plate-out is traditionally referred to as “liftoff”, see Fig.2.

In addition to liftoff, some volatile fission products that are chemisorbed on primary surfaces can desorb if there is temperature increase at the surfaces.

Following the failure of the TRISO particles, the fission products will be released into the reactor primary system. The fission product transport codes would calculate the transport of fission products in the primary system. If the accident involves a pipe break, the fission products will be released into the containment. If the containment is a low-leakage containment, most of the fission products may be contained inside the containment. If the containment is a vented containment, some of the fission products within the containment can be released to the environment.

### 2.4.1 Air Ingress

In some accidents there is a probability of air ingress. The air ingress may lead to graphite oxidation and “burning” (which is usually self sustained). The amount of oxidation depends on the amount of air ingress through a break. This depends on the containment (or confinement) design, the size and location of the break, penetration of the air into the core and the amount of the air penetrated. The oxidation of graphite is of primary importance with respect to the structural integrity of the graphite. The surface penetration on the graphite will depend on the nature of the oxidant, its concentration or partial pressure, the temperature, the type of graphite and the time. Decrements in the strength and structural integrity of the graphite core and support components will depend on the depth of penetration.

When the oxidant reaches the site of the fuel particles, oxidant-fuel interactions are possible in two ways. In the first way, some particles may have failed during the course of the preceding irradiation or fabrication process in such a way that the fuel kernels are exposed to the particle exterior. In this case, the oxidant can directly reach the fuel, induce oxidation, and change the structure of the grains of fuel. The consequence is highly enhanced release of fission products. The second way is the oxidation of the outer pyrocarbon coating, the SiC coating, and inner pyrocarbon and buffer coatings. Loss of structural integrity in these layers leads to a larger fission product release than would happen in the first way. Hence, oxidation plays an important role in governing the fuel performance.

This would suggest that thermal hydraulic codes should have the capability of calculating the air penetration through the break, piping and into the core. In addition to a thermal hydraulic code, there should be a code module to predict graphite oxidation, degradation of the graphite structure, chemical processes and release of fission products.

### 2.4.2 Water Ingress

In some accidents there is a probability of water ingress. Water, as it is transported, would heat up and it is assumed that it would be in vapor form as it arrives at the core. The interaction of water vapor with exposed fuel particles depends on the partial pressure and temperature of the vapor. The increase in fission product release, mostly in form of isotopes of Kr and Xe, occurs during the period of exposure to water vapor. In contrast, if the particles are intact: i. e., the fuel particles are not exposed because of manufacturing defects or irradiation, they are not damaged by the presence of water vapor. Hence, the amount of increase in fission products would depend on manufacturing defects and failures due to irradiation as well as duration of water vapor exposure, its partial pressure and temperature.

Another effect of the water vapor ingress is the fission product “washoff” of the plated out fission products, see Fig. 2. If water ingress has occurred during a pipe break, the washoff increases the fission product release during the accident, Ref. 2.

This would suggest that thermal hydraulic codes should have the capability of calculating water vapor penetration into the core. In addition to this capability, there should be a code module to predict fission product increase based on manufacturing defects and irradiation failures given the duration of the presence of water vapor, its partial pressure and temperature by the thermal hydraulic code.

## 3.0 DISCUSSION OF COMPUTER CODE NEEDS

### 3.1 TRAC-M

TRAC-M is a system transient analysis code for LWRs. Some changes are being made to add capabilities to calculate transients and accidents in gas cooled reactors particularly in PBMRs. A code requirements document identifies requirements to calculate transients and accidents for a Pebble Bed Reactor, Ref. 4. A short summary of requirements are given below:

1. Helium fluid properties have been added as part of ATW updates. Helium equation of state properties are in terms of correlations. The correctness of these correlations will be assessed.

2. Pebble bed will be treated as a porous medium. 3-D thermal hydraulic phenomena will be analyzed using the TRAC-M vessel component. This will permit calculation of natural circulation within the pebble bed. The user has the capability to model the pebble bed using a nodalization scheme for different axial levels, radial rings and angular sectors. This will give the average values of parameters such as temperatures, pressures and flows at different regions of the pebble bed. In order to accomplish core heat transfer capability, new correlations of effective thermal conductivity for a porous medium and new correlations for the pressure drop in a porous medium are being added. The effective thermal conductivity correlations simulate the effects of conductivity between the pebbles where they contact each other and also the radiative heat transfer from pebble to pebble across the core. Hence, heat transfer from the core to the walls of the vessel and through the passive cooling system around the vessel to the soil can be calculated.

3. Nuclear graphite material properties dependent on temperature and exposure will be added. It is expected that thermal conductivity in the pebbles will be uniform and isotropic; however, in the reflector region the thermal conductivity is unisotropic. Unisotropic properties are important to simulate the passive conduction cooling.

4. Gas turbine and compressor components will be added.

### 3.2 Fuel Element Performance Code / Module

Section 2 discusses where uncertainties are, in calculations of fission product release performed by GA. In order to predict fission product release, it is necessary to develop a computer code or module. This module or code will run with TRAC-M in parallel. This module should predict the fission product release addressing all uncertainties discussed in Section 2 including air and water ingress events.

Similar codes or modules have been developed and used by the industry. For example, GA developed and used SORS code to calculate fission product release from TRISO particles together with their thermal hydraulic code, CORCON, which provided input temperatures to the SORS code. These two codes were used in a coupled manner. GA has also developed OXIDE3 code for analysis of steam and air ingress events. Similarly, Research Center Julich developed and used REACT and THERMIX codes. The THERMIX code is a thermal hydraulic code while REACT is a code simulating chemical processes during the air ingress accident. They are run coupled to predict the fission product release during the course of water ingress events. JAERI has developed GRACE and OXIDE-3F codes for water and air ingress events. It appears that both vendors and research institutes developed their own codes to predict fission product release with / without air and water ingress events.

The phenomena occurring during fission product release from TRISO particles are very different form those occurring in LWRS. Modification of a code used for LWRs, such as MELCOR, may not be suitable and cost-effective to simulate fission product release in gas cooled reactors. It is strongly recommended that a separate fission product release computer module be developed. This module would be run coupled with the TRAC-M code which would provide input temperatures and input partial pressures for air and water vapor to the module.

### 3. Dust Transport & Deposition & Liftoff Code Module

As discussed in Section 2.3, dust is produced as part of normal operations in PBMRs. The dust particles are transported and deposited at certain locations in the primary system. When a depressurization event or a pipe break accident occurs, the deposited dust may be lifted off and be carried away into the reactor building and/or to environment. As discussed in Section 2.3, this dust is radioactive and contributes to initial fission product release. Amount of the release depends on the amount of deposition and liftoff. If there is also water ingress, some of the plated out radionuclides may be washed off and carried away into the reactor building and/or to environment.

It is necessary to develop a computer code module to simulate dust transport, deposition and liftoff together with plateout and washoff phenomena. In this area, the data are relatively sparse since amount of dust produced is a function of reactor operations. Some filtering systems were designed in VAMPYR-II tests conducted using a bypass flow from the AVR reactor, to measure efficacy of the filters to reduce dust and plateout, Ref. 2. There were significant differences between measured plateout and computer code predictions (SPATRA and PATRAS) , Ref. 1.

This is an area where very little data exist. New test data will be necessary. It is also necessary to develop a computer code module based on existing data. Perhaps initially conservative models can be used and as we obtain more data, these models can be refined and margins be reduced.

**Conclusions**

Reviews of some regulatory issues concerning TRISO particles and state of the research performed in different countries are performed. These reviews showed the issues and areas where uncertainties exist and where further research efforts should be directed. The regulatory issues have been resolved in a conservative way. The uncertainties in assessment of integrity of TRISO particles have contributed to the selection of a low leakage type of containment for the Ft. StVrain plant, Ref. 5.

In terms of thermal hydraulic codes, TRAC-M is suitable for modifications. Limited amount of modifications will be required to predict the phenomena in gas cooled reactors. However, major efforts are required to develop a fuel element module for prediction of fission product release from TRISO particles in gas cooled reactors. A second major module for prediction of dust transport, its deposition and liftoff and fission product plateout and washoff is necessary.

It is necessary to develop detailed requirements documents for these modules after the PIRT program on TRISO particles is completed.

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