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TÜV HANNOVER

HTR Two-Module Power Plant Facility

Safety-technological Design Evaluation

Prepared under contract with the
German Federal Ministry for Research and Technology

October 1989

Safety-technological Design Evaluation of the HTR Two-Module Power Plant Facility

October 1989

TÜV Hannover, e.V.

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Introduction

In compliance with § 7a of the AtG (German Federal Nuclear Energy Act), the companies Siemens/KWU and Interatom submitted an application, dated April 29th,1987, for a provisional, site-independent approval of the HTR-2 (high temperature reactor) module power plant plan to the Minister of Environment of Niedersachsen (MU-Nds – Niedersächsischer Umweltminister). The High-Temperature Module Planning Company (HMP – Hochtemperatur-Modul Planungsgesellschaft) joined this application with a letter dated June 4th, 1987.

With a letter dated May 29th, 1987, the Ministry of Environment of Niedersachsen contracted the TÜV Hannover e.V. to conduct a technical safety assessment and subcontracted the TÜV Rheinland e.V. to participate in this assessment. This subcontract allowed the incorporation of experience gained during the monitoring and licensing proceedings of the AVR pilot reactor in Jülich. Furthermore, our assessment also took into account the evaluation results and operational experience of the THTR-300 in Hamm Uentrop.

Fire protection and safety aspects regarding breakdown measures or other third party effects were excluded from this assessment. Also contracted by the Ministry of Environment of Niedersachsen, these aspects are subject of a separate assessment.

The application for a provisional approval was withdrawn in April 1989 and the licensing proceedings were terminated by the Ministry of Environment of Niedersachsen.

Consequently, the contract with TÜV Hannover e.V. to conduct a technical safety assessment was cancelled by the Ministry of Environment of Niedersachsen with a letter dated April 12th, 1989, effective April 30th, 1989. At that time, the assessment was already well underway and much interest in the completion of the project was present. In a letter dated March 21st, 1989, the TÜV Hannover e.V. offered to the German Federal Ministry for Research and Technology (BMFT) to complete the technical safety assessment of the plan with no reference to any particular approval procedure within the scope of a research and development contract and to document the results in the form of an expert statement. The pertinent research and development project has been submitted on March 30th, 1989. In May 1989, the German Federal Ministry for Research and Technology contracted the TÜV Hannover e.V. to conduct this assessment.

The changed contract conditions have caused us to abandon our original strategy to divide the assessment into two parts. It had been planned to carry out a "conceptual assessment" as the basis for the "preliminary positive opinion in respect to the construction and operation of the power plant" as required by § 18 of the Regulations under the German Federal Nuclear Energy Act (ATVfV – Atomrechtliche Verfahrensordnung), in which the following aspects were to be examined and documented as to whether:

-Preventive measures according to the current state of science and technology against damages during construction and operation of the HTR Module of the power plant can be met, and

-Protective measures necessary for pollution prevention of water, air and soil can be ensured.

In addition, an "advisory opinion pertaining to the provisional approval" regarding the subject matter of the application was to be provided. In Chapter 2 of the Safety Report /U 1/, the applicant (Siemens / Interatom / HMP) has separately documented and codified the technical construction criteria for which provisional approval was sought.

Since our assessment no longer relates to a specific step in the approval proceedings and is not subject to legal procedures in compliance with nuclear energy regulations, we combined the originally considered two partial evaluations into this "Technical Safety Plan Assessment". This assessment has been divided into sections according to the Directive 13 of the "Standard Organization of TÜV /GRS Assessments of Nuclear Power Plants". This organization had been created for the use in assessments of nuclear power plants with reactors using the pressurized water or boiling water systems and, therefore, had to be somewhat modified for the HTR module power plant. In addition, we have taken into account requirements of the "Framework Guidelines Pertaining to the Organization of Expert Opinions /L 19/.

The application for a provisional approval of the power plant plan had been filed site-independent. Thus, in Chapter 1 "Site", data provided in the technical safety statement with respect to the site characteristics were evaluated for their completeness and plausibility only, not relating to any specific location. Our statements with respect to the technical implementation of the power plant are generally valid under the assumption that a specific location will not introduce deviating boundary conditions. *E.g.*, the evaluation of the emergency power supply plan is based on the reliability of third party electricity supply as available in the Federal Republic of Germany.

In this technical safety assessment and in accordance with our established procedures, descriptive sections as well as sections relating to specific facts have been emphasized by indentation. These evaluating sections list a number of suggestions, instructions and requirements regarding the realization of technical safety objectives. Also codified are important boundary conditions on which our statements are based. We have foregone to specifically emphasize or list these suggestions and requirements separately, because the plan is not codified with respect to regulatory proceedings. These annotations, however, are relevant for the further construction planning of the HTR-2 module power plants, as their observance is prerequisite for the acceptability from a technical safety viewpoint.

1. Site Characteristics

1.1 Site-independent Layout Design

The HTR-2 module power plant described in the Safety Report /U 1/ can be used for combined generation of electricity, process steam and/or district heating as well as for injection steam generation in tertiary oil production. The site resulting from this spectrum of applications is an industrial site. The layout design of the plant takes into account certain assumptions about the site characteristics, which are assumed to be representative for a large group of potential locations.

Experiences with other nuclear energy facilities provide information, which characteristics of a site can affect the layout design and operation of the facility. Therefore, we have examined whether the assumptions outlined in the Safety Report with respect to anticipated location characteristics are complete, their selection is sound, and whether they meet the requirements of the nuclear power regulations of the Federal Republic of Germany.

1.2 Population Distribution

The population distribution surrounding the site will correspond to that of industrially intensively used areas. According to the specifications of the Safety Report, the HTR-2 module power plant is designed so that any release of radiation during conventional operation as well as in case of all layout design accidents will be below the maximum values of the German Federal Radiation Protection Ordinance (StrlSchV – Strahlenschutzverordnung) so that the directive concerning radiation minimization can be upheld /L 2/. Evidence will be furnished for a reference person at the most concentrated emission point and is, thus, independent of the actual population distribution.

The German Federal Radiation Protection Ordinance does not have any maximum values for a collective dose. The calculation of the radiological effect of a nuclear power plant, thus, is done independent of the actual population distribution. The actual population distribution at a specific location, thus, does not place any additional requirements on the layout design of the planned facility. However, information about the population distribution will be necessary for planning of emergency protection measures at an actual location.

1.3 Land and Water Use

Regarding land and water use, the conditions presumed in the Safety Report are such that the details given in § 45 StrlSchV /L 13/ can be used for the determination of radiation exposure from food chains.

> Land and water use at an actual site does not pose any additional requirements regarding the layout design of the facility. However, it needs to be examined whether site-dependent exposure paths, not accounted for by the calculation model subject to § 45 StrlSchV, could contribute to a significant extent to the radiation exposure.

1.4 Meteorology and Hydrology

The provisions of § 45 StrlSchV have been applied for the detection of possible radiation exposure of persons in the surrounding area to radioactive materials emitted by waste air during conventional operation. In addition to the parameters stated in the above calculation, meteorological data from the Safety Report have been used for a representative location of a HTR-2 module power plant to describe the dispersion of radioactive materials emitted into the atmosphere. According to the calculation models of § 45 StrlSchV, these meteorological data provide following long-term dispersion factors for the calculation of radiation exposure:

<u>Emission Source: Ventilation Stack</u>

Chi_{gamma}	$= 6.0 \times 10^{-3}$ seconds/cubic meters	(100 meters from the ventilation stack)
Chi_{conc}	$= 6.5 \times 10^{-7}$ seconds/cubic meters	(100 meters from the ventilation stack)
Chi_{conc}	$= 1.7 \times 10^{-6}$ seconds/cubic meters	(270 meters from the ventilation stack)

<u>Emission Source: Machine Building Roof</u>

Chi_{conc}	$= 4.1 \times 10^{-5}$ seconds/cubic meters	(100 meters from the ventilation stack)
Chi_{conc}	$= 9.7 \times 10^{-6}$ seconds/cubic meters	(270 meters from the ventilation stack)

The Safety Report also makes assumptions about the average wind speed and average precipitation volume. These data are also needed for calculation of the potential radiation exposure.

The calculation of radiation exposure for persons through radioactive emissions from wastewater was based on general information from the Safety Report. It was assumed that the average wastewater outflow is completely mixed and has an average capacity of 30 cubic meters/second at the outfall.

> In our opinion, the assumptions in the Safety Report concerning the meteorological and hydrological data for calculation of the radiological impact of the HTR-2 module power plant are chosen appropriately. To our knowledge, there are a number of locations in the Federal Republic of Germany at which the meteorological data result in similar long-term dispersion factors corresponding to the values in the Safety Report and where an outfall with an average water flow of more than 30 cubic meters/second is available as well.
>
> Other meteorological phenomena, such as lightning, wind, snow, and ice-formation are taken into account consistent with conditions in the Federal Republic of Germany and according to information from the Safety Report. However, in our opinion, they will not have any impact on the layout design of the HTR-2 module of the power plant facility.

1.5 Floods

According to general information from the Safety Report, the following anti-flood measures are taken into account:

- The external walls up to the design flood water level are manufactured from water-impermeable concrete subject to DIN 1045 (German Industry Norm – Deutsche Industrienorm) /L 159/.
- The stability is demonstrated for the design flood water condition
- All entries to the protected buildings are situated above the design flood water level.
- All pipes and cable ducts below the design flood water level are waterproof.

The design floodwater will be determined from the hydrological conditions of an actual location. Depending on the location characteristics, additional measures for floodwater protection, such as earth banks, may be necessary. These, however, do not have any effect on the design of the facility.

1.6 Explosive, Poisonous and Corrosive Gases

According to the Safety Report, protection against penetration of gases that could build up an explosive atmosphere is provided for the reactor building and the emergency control point. The layout of the ventilation facility inside the auxiliary reactor building also functions as its protection against penetration of explosive gases. The air intake vents of the auxiliary reactor building facility are equipped with throttle valves, which will be closed in case of occurrence of explosive gases. Gas alarm occurs via the gas alarm setup for explosive gases or through administrative measures.

Poisonous gases do not lead to a disturbance in the facility, but they may cause dropout of plant personnel. In case of alarm, air ventilation may be closed in the reactor building, auxiliary reactor buildings, as well as in the switching station, and emergency power supply building. Independent thereof, respirators with pressured air supply are to be stored in the control room for the personnel to ensure that the breathing air supply is independent of the ambient air.

Corrosive gases will not have any damaging short-term effects on the facility. Shutdown of the facility and inspections are, thus, always possible. Detection measures are not necessary. In case of alarm, the ventilation shutdown can be activated in the reactor building, auxiliary reactor buildings, as well as in the switching station and emergency power supply building.

> The layout design of the facility provides for site-independent protective measures against the intake of explosive, poisonous or corrosive gases. The extent of these measures depends on the predicted impacts. Ambient conditions at a specific location may result in requirements regarding protective measures against specific types of gases. This however does not result in any impact on the layout design of the facility.

1.7 Shock Waves

According to the Safety Report, those parts of the HTR-2 module power plant facilities that are relevant from a technical safety point of view, are constructed so that they can withstand the impact of explosion shock waves. Unless required differently by particular features of the location, construction against shock waves resulting from chemical explosions is based on the following assumptions regarding their progression:

- Increase of excess pressure in 0.1 seconds to 0.45 bar,
- Decrease in 0.1 seconds to 0.3 bar,
- All-side quasi-static load of 0.3 bar in the next 0.8 seconds.

The shock wave may arrive from any direction.

The load situation of an explosion shock wave is usually considered to belong to the residual risks because of its low probability of occurrence. Site-independently, nuclear power plants in the Federal Republic of Germany are designed to minimize this residual risk from the impact of a shock wave resulting from chemical reactions. The load assumptions that have to be met are specified in the directive /L 17/ and correspond with the assumptions specified in the Safety Report.

Load situations caused by shock waves, higher than those that are specified in the site-independent layout design, may occur depending on the properties of the explosive material. Protection against such shock waves is typically assured not only by construction measures but additionally by adequate safety distances to possible explosion points. When selecting a specific location – especially close to industrial plants – it should be evaluated whether safety distances to potential explosion points according to the requirements of the regulation are available or alternatively, whether stricter construction requirements have been into account.

1.8 Earthquake

The Safety Report defines two types of earthquakes for the layout design of the HTR-2 module power plant from assessment point of view – the safety earthquake (SE) and the design earthquake (DE) – and also specifies their respective seismic engineering indicators.

The input rates lambda of these dimensioning earthquakes and their respective intensity I according to the MSK scale representative for a large group of potential locations is defined as follows:

Safety earthquake (SE) Lambda (larger than I) = 10^{-5} /year
Design earthquake (DE) Lambda (larger than I) = 5 x 10^{-3} /year

Safety earthquake I = VII – VIII (7.25)
Design earthquake I = V – VI (5.5)

In addition to intensity, the definition of the seismic engineering indicators should also take into account the subsoil conditions of the location. The Safety Report considers two types of soil, A (soft subsoil) and M (medium hard subsoil). The procedure regarding the determination of realistic free field seismic spectra for a location in the Federal Republic of Germany is described in the Safety Report.

For the safety earthquake, the Safety Report specifies / U 1-1/ the following boundaries for response spectra for the horizontal direction components (free field, 50 percent fractiles, damping 5 percent):

Soil type A

Frequency (Hz)	0.5	1.5	5.0	20
Acceleration (meters/seconds^2)	0.28	1.6	1.6	0.81

Soil type M

Frequency (Hz)	0.5	3.0	7.0	25
Acceleration (meters/seconds^2)	0.20	3.3	3.3	1.4

The boundaries of the response spectra of the design earthquake have been determined as follows:

Soil type A

Frequency (Hz)	0.5	2.5	6.5	20
Acceleration (meters/seconds^2)	0.05	0.52	0.52	0.28

Soil type M

Frequency (Hz)	0.5	3.5	7.5	25
Acceleration (meters/seconds^2)	0.04	0.80	0.80	0.38

The acceleration values of the assessment spectra for the vertical direction components are defined as ⅔ of the values for the horizontal direction components. Duration of a strong earthquake is specified at 5 seconds for soil type A and 3 seconds for soil type M.

We judge the determination of seismic load assumptions in the Safety Report to be a consistent procedure for the realistic description of earthquake impact loads. The necessary site-specific seismic expert opinion should make use of this procedure in consideration of the present deterministic method in order to specify the final seismic engineering indicators without any loss of safety (see also Section 5.6).

The following evaluates the seismic assumptions described in the Safety Report and conceptually assessed in the layout design evaluation.

The documents /L 91/ and /L 92/ which describe the utilized procedures for determination of the load assumptions serve as a basis for the assessment.

The choice of intensity and subsoil corresponds in all intents ands purposes to conditions of conceivable locations within the Federal Republic of Germany. At any specific location, the intensity and other macro-seismic indicators (magnitude, distance from epicenter) as well as the soil indicators need to be determined on basis of a site-specific scientific expert opinion regarding its seismology and subsoil conditions.

Assessment spectra for the horizontal soil acceleration (direction component) are given /L 91/ as statistically determined average value spectra (50 percent spectra). These spectra are valid for the average intensity I_m within a class (*i.e.* $I_m = 7.5$ for the class $I = 7$ to 8) and are consequently scaled to the location intensity by a factor.

In comparison with /L 91/ it appears that the Safety Report derived assessment spectra for safety earthquakes, soil type M, somewhat conservatively (it corresponds to approximately $I = 7.4$ instead of $I = 7.25$). The other spectra for safety earthquakes, soil type A, and design earthquakes, soil type M and A, correspond to the data in /L 91/ after scaling factors have been taken into account.

The range of the vertical assessment spectra with ⅔ of the value for the horizontal acceleration components corresponds to the state of the technology.

The assumption of a strong earthquake duration of 5 seconds in case of soil type A and 3 seconds in case of soil type M follows the recommendations in /L 91/ and/or /L 92/.

In the same way as macro seismic and soil dynamic indicators, the engineering seismic indicators of the Safety Report ultimately have to be checked with the results of a site-specific seismic expert opinion.

The acceleration impact caused by earthquakes on the reactor core of the HTR-2 module power plant facility result in an increase of reactivity due to increased density of the fuel element fill. According to information in the Safety Report, the layout design for the fuel core has been based on the maximum horizontal acceleration due to earthquakes of 0.5 g and a duration of 6 seconds. Aside from the site-specific earthquake load assumptions, the assessment of this design layout for the fuel core also takes into account the response spectrum of the fuel core. The reactivity increase due to earthquake impacts on the reactor fuel core is commented on in Section 5.2.6.

The assessment of the seismic conditions for a specific site needs to review whether the presumed earthquake loads for the fuel core take site-specific conditions adequately into consideration.

1.9 Airplane Crash

The risk of a crash of a fast flying military airplane is approximately the same in all conceivable locations within the Federal Republic of Germany. The parts of the HTR-2 module power plant facility are, thus, designed according to site-independent general information in the Safety Report regarding impacts due to an airplane crash. The following load assumptions are assumed:

- impact-load time progression (in case of a crash against a rigid wall)

Impact time (milliseconds)	Impact load (MN)
0	0
10	55
30	55
40	110
50	110
70	0

- Impact surface

  The impact surface is assumed to be round with an area of 7 square meters.

- The impact angle is considered to be vertical to the tangential level of the impact point.

An airplane crash event onto a nuclear energy facility is classified as a residual risk due to the small probability of its occurrence and, therefore, is not listed among the failure cases for which compliance with the planning guidelines of § 28 section 3 StrlSchV /L 2/ has to be demonstrated. However, the design layout for new nuclear power plant facilities in the Federal Republic of Germany, has to be site-independently devised to minimize the residual risk due an airplane crash. The load assumptions given in the Safety Report reflect the requirements of the RSK guideline for pressure water reactors /L 10/.

1.10 Preexisting Radiological Contamination

The assessment of the suitability of a site must ensure that the limits of the § 45 StrSchV concerning admissible radiation exposure in the surrounding area will not be exceeded, even when taking into the preexisting radiological contamination of the location due to other nuclear facilities. The Safety Report does not mention the preexisting radiological contamination. The assessment of a specific location must determine the preexisting radiological contamination in the air, water and soil due to other nuclear facilities and should take this information into account when calculating the predicted radiation exposure.

1.11 Summary Assessment

The layout design of the HTR module power plant takes into account load assumptions and design requirements of the Federal Republic of Germany that need to be considered either site-independent or that result from conditions at many potential sites. Because the assumptions in the Safety Report cannot cover the entire possible spectrum of site conditions, the selection of a specific location requires verification that the actual site conditions are adequately considered in the design layout of the facility.

2. Power Plant Facility

2.1 Subject of the Evaluation

The HTR-2 module power plant facility is described in the Brief Description /U 2.1–1/, the Safety Report /U 1/ and in additional documents /U 2.3–1/ to /U 5.8–7/.

Within the framework of the Nuclear Energy Act approval proceedings, the Safety Report plays the role to permit third parties (the public) to evaluate whether their rights could be violated by the effects of the facility or its operation. This is why the Safety Report introduces and explains the concept (the basic design characteristics), the safety-technological principles of the design and the function of the facility including its operation and safety systems and it describes the effects of the facility and its operation including the effects of breakdowns in the sense of § 28, section 3, sentence 4 of the Radiation Protection Regulation /L 2/. With regard to its task and the approval proceedings, the descriptions of the Safety Report are arranged in such a way to give a general overview and to describe the preventive measures that § 7, section 2, No. 3 of the German Federal Nuclear Energy Act specifies as necessary to comply with the approval proceedings. At the beginning of our evaluation, the Safety Report was available in its draft version (4/87).

The Safety Report as a single document is not sufficient to deal with all safety-technological aspects of the design of the facility in detail. This is why the applicant has submitted additional documents in the form of complementary plans, drawings, functional and structural descriptions, etc. These documents together with the Safety Report were used as the basis for a series of expert meetings that were held in the second half of 1987 between us and the applicant after the order had been placed and it related to important parts of the facility as well as the safety-technological aspects. In this expert meeting, the applicant gave us additional explanations and we performed and discussed the first safety-technological evaluations.

After the end of this first evaluation stage in February 1988, TÜV Hannover, e.V. issued a list of required documents containing information necessary for further stages of the evaluation. /U 2.1– 2/. At a general meeting at the beginning of March we explained these requirements while several requirements for technological changes resulting from the first evaluation stage were described and justified.

The applicant revised the presented documents on the basis of these findings and an additional meeting held in the first half of 1988 concerning safety-technological conditions (*e.g.*, the concept of stand-by power supply, the closedown concept, mechanical breakdown analyses of the pressure tank unit, etc.) and submitted additional and complementary technical documents. The draft of the Safety Report was revised as well (version 8/88 of November 1988).

By December 1988, all essential documents were available, however, until September 1989 additional documents were submitted that were taken into account for the safety-technological evaluation on hand if it was still possible.

The subject of our evaluation is described unambiguously in the documents that are quoted in the following sections. The references to documents submitted by the applicant are marked with the letter "U" preceding the number and they are summarized in Section 7.1.

We are aware of the fact that the construction design of the facility is currently still proceeding and that it will be further substantiated within the framework of the presented position and the report on hand, which evaluate the concept of the facility. A further step will be to assess whether our recommendation for the construction design contained in this report have been observed and whether the boundary conditions and prerequisites mentioned in our statements for the construction design have been maintained.

2.2 Design Requirements

The legal and technological regulations that are to be observed in the course of approval proceedings in the Federal Republic of Germany are organized hierarchically and thus their level of detailed approach varies accordingly.

The German Federal Nuclear Energy Act /L1/ and the Radiation Protection Regulation /L2/ represent the general legal base for the planning, building, operation and closedown of nuclear facilities as well as their supply and waste removal and thus also provide the framework for the safety-technological evaluation. They do not refer to designs of the facilities or detailed technological solutions; the requirements of these regulations must always be complied with.

Below the legal regulations there are a set rules and regulations that refer more closely to technological concepts and which were compiled parallel to the development of nuclear technology to provide unified evaluation criteria and to describe the state-of-the-art technology. This holds especially true for technical regulations. However, the regulations mostly refer to light-water reactors. This is why the regulations for nuclear technology are largely shaped by the physical and conceptual characteristics of light-water reactors and particularly by a pressure water reactor with a high power load nuclear design and they cannot be used for other types of reactors indiscriminately.

Gas-cooled high temperature reactors are characterized by distinct characteristics of inherent safety regarding reactivity malfunctions and obstruction of heat removal. This is especially true in the case of a low power-per-reactor unit and with lower power density in the reactor core. This is why it would be inadequate to project requirements from the technical regulations of the light-water reactor onto the HTR module without any reflection, particularly when taking into account that in this case a reactor concept was created under consequent utilization of the safety properties of a gas-cooled small-power reactor with the aim to limit fuel element temperatures even in the case of a total failure of all active cooling systems and the loss of coolant to such an extent that no significant escape of radioactive fission products from the fuel elements could occur. Such a claim goes beyond the criteria that have been used so far for high-temperature reactors. For the use of the nuclear-technological regulation system this means that even the conceptual regulation system for high-temperature reactors needs to be specifically adapted and transferred to the conditions of an HTR module reactor.

In our safety-technological evaluation we used the following super ordinate rules and regulations and we examined whether the observation of their requirements is transferable to an HTR module:

- Safety Criteria for Nuclear Power Plants /L 6/

  The requirements specified in the criteria are so fundamental that they can be used directly with some small exceptions. However, this does not mean that technological solutions resulting from the long-term practice of using the safety criteria can be used indiscriminately but that the safety-technological significance of the facilities and their functions is derived from the facility design in advance according to the specifications of the criteria.

- Safety Criteria for Power Generating Facilities with Gas-cooled High-temperature Reactors – Draft 1980 /L 7/.

  This draft is closely related to the Safety Criteria for Nuclear Power Plants /L6/ with an adaptation to the specifics of gas-cooled high-temperature reactors. To be able to use these criteria for the HTR module reactor, the design-specific safety-technological implications of the mentioned facilities and their functions have to be considered.

- Concept of Individual Failures – Principles for the Use of the Individual Failure Criterion /L 8/ and Safety Criterion Interpretation 2.3 /L 9/

  From the number of BMI interpretations of safety criteria for nuclear power plants we have used the above-mentioned interpretation to illustrate the requirements mentioned in the safety criteria. The previously mentioned prerequisites for safety criteria apply, when using them for the HTR module reactor.

- RSK Guidelines for Pressure Water Reactors /L 10/
- RSK Guidelines – "Basic Safety" /L 12/
- Breakdown Guidelines /L 11/

  Unlike the safety criteria and their corresponding interpretations, these guidelines are strongly focused on the concept of pressure water reactors and certain technological designs. However, some sub-areas can be used design-independently (*e.g.*, application of basic safety, determination of loads resulting from a plane crash, limitations of design breakdowns of hypothetical processes). For the rest, we have used the guidelines as a collection of observations to check the completeness of our evaluation procedures.

In addition we have used the following, mostly conceptually independent guidelines for our evaluation:

- Calculation Basis for § 45 of the Radiation Protection Regulation /L 13/,
- Calculation Basis for the Breakdown Guidelines (StrlSchV § 28.3) /L 14/,
- Planning Provisions Guideline /L 15/,
- Guideline to Emissions and Immissions Monitoring of Nuclear Facilities /L 16/
- Guideline to Measures Against Shock Waves /L 17/,
- IAEA-Safety Guide No. 50-SG-D1
  Safety Functions and Component Classification /L 21/ for BWR, DWR and PTR,
- International Commission on Radiological Protection /L 22/, ICRP Publication 26, Oxford 1977, /L 22/,
- International Commission on Radiological Protection /L 23/, ICRP Publication 30, Oxford 1978–1985 /L 23/,
- Guideline for Waste Water Discharge from Nuclear Power Plants with an LWR into Water Courses (LAWA) /L 24/,
- SSK, Appendix 9 to the Minutes of the 65th Meeting held on April 17th and 18th, 1986 /L 25/
  Opinion of the Radiation Protection Committee Regarding the Modification of the Existing Calculation Basis Used to Determine the Radiation Exposure of Nuclear Fuel Cycle Installations.

Obviously, technical regulations are much more focused on facility design and operating conditions of systems and components than super ordinate guidelines. Based on the HTR module concept, we have therefore used individual technical regulations for nuclear power or conventional power or parts thereof from the regulation sets, *e.g.,*:

- KTA Rules,
- DIN, ISO,
- ASME-Code,
- ANS-Rules,
- AD-Material Sheets,
- VdTÜV-Material Sheets,
- TRD (Technical Rules – Steam Boilers),
- TRB (Technical Rules – Pressure Tank),
- Steam Boiler Regulation,
- Pressure Tank Regulation.

In as much as they are representative for the state-of-the art technology and applicable to the HTR module facility or if their observation is necessary regardless of the design (*e.g.*, the Pressure Container Regulation). It was not part of our intention to test and classify the applicability of the entire set of regulations for nuclear power and conventional power to the HTR module within the scope of this safety-technological evaluation. Our set of utilized regulations and standards is thus an example only and not comprehensive /L 26 to L 74 and L 159 to L 163/.

The German Federal Nuclear Energy Act requires a safety-technological evaluation of a nuclear facility with respect to necessary damage prevention according to the state-of-the-art of science and technology. This aspect is particularly important in the case of a prototype facility. This is why we have in particular used existing evaluations of HTR facilities based on the pebble bed principle /L 76–L 90/ and we have taken into account the current status of scientific publications /L 91–L 158/. Again, we do not claim comprehensiveness in regards to the evaluation of existing sources. However, it is typical for the evaluation of a prototype facility with a new design that this part is the largest share of the literature we have used and quoted.

The literature we have used for the evaluation (regulations, table compilations, publications, etc.) is always referred to in the text and marked with the letter "L" and consecutively numbered. All these quotations are summarized in Section 7.2.

2.3 Nuclear Technical Classification

As required by the safety criteria for nuclear power plants /L 6/, criterion 2.1 and the draft of the safety criteria for gas-cooled high-temperature-reactors /L 7/, the quality of all parts of the nuclear power plant must be oriented towards their safety-technological importance. In compliance with this requirement the

- Pressure and activity conducting systems,
- Ventilation facilities,
- Lifting machinery and cranes,
- Steel structural parts

are arranged in Section 2.4.1 of the Safety Report /U 1/ and supplementary documentation /2.3–1/ in the quality categories depending on their respective safety-technological importance. Quality indicators and quality tests are described in Section 2.4.2 of the Safety Report, on which the proper design should be based, corresponding to their safety-technological importance.

The electro-technical and control technology systems and components are not classified in the Safety Report. However, a classification is provided nonetheless /U 2.3–2/. As this classification is not subject of the contracted preliminary report, we do not go into details.

The installation of components is, in general, according to the Safety Report, not to be classified. Unlike the structural parts of the classified components, these installations are components with their own KKS-designation. It is planned that the requirements for these structural parts according to their safety-technological function will be given in a separate specification /U 1/.

However, this is not planned within the framework of the concept planning, but rather during implementation planning.

In our opinion the applicant's intention to classify the facility parts, as early as in the concept design phase, is an accurate and reasonable measure to obtain the proper safety-technological facility design, particularly considering the prototype nature of the HTR-Module-Power plant facility. However, based on the present state of planning we consider the final assessment the mandatory assignment of quality classification and quality ensuring measures in as early as in this step to be too soon. For this reason, we have evaluated the presented categories and their basic criteria just in terms of the proper consideration of the various safety-technological importances of the facility parts regarding integrity and function. To illustrate these circumstances more clearly, we used the term "safety category" instead of the term "quality category" used by the applicant.

Based on the safety-technological importance (safety categories), we generally consider the levels (quality categories with the respective quality ensuring measures) given in the Safety Report as adequate to determine the inherent safety properties of the HTR-Module-Power plant facility and to ensure the required quality of the facility parts.

2.3.1 Pressure and Activity Conducting Systems

According to the Safety Report, the pressure and activity conducting systems are classified based on the integrity of the pressure conducting system walls.

This means that for these systems, the applicant applies the safety-technological requirements regarding the integrity of the pressure conducting system walls within these systems as the only criterion for the safety-technological classification of the system and the components.

The applicant proposed the following categories and criteria:

MK1 – when a loss of integrity could result in exceeding the dose limits according to § 28 paragraph 3 of StrlSchV,

– when a primary leakage could occur that cannot be sealed off – irrespective of radiation exposure

MK2a – when the systems for the control of components and structures are needed for secondary heat removal or when they are needed for module shutdown,

MK2b – when the systems reduce the release of radioactivity to values below the specified limits in accordance with the designed operation,

NNK – when the systems are not classified as MK1 or MK2.

The applicant assigned the systems to individual categories in the Safety Report and the supplementary documentation /U 2.3–1/. We also give our opinion on this in the following:

We assess the safety-technological importance of the pressure and activity conducting systems based on the safety functions given in the Safety Criterion Interpretation /L 9/. These are similarly specified in the IAEA-classification rules /L 21/, and here they are applied to the classification of the pressure conducting walls of pressure and activity conducting systems. According to that, we consider the safety-technological functions given in the decision criteria to be exhaustive. However, systems and components whose failure could affect the function of the safety-technological parts as well as the safety-technological parts of the auxiliary and accessory systems are not mentioned as a nuclear technology classification criterion. This will be discussed in more detail later on.

The category MK1 includes systems whose loss of integrity could cause a primary leak that cannot be sealed off or an inadmissible radiation exposure. This division is sensible and in general complies with the present approval practice.

The classification of systems and system areas in the MK1 category as given in the supplementary documentation /U 2.3–1/ is generally congruent with that given in the Safety Report and is considered exhaustive based on the present planning stage. It should be evaluated during the detailed implementation planning whether perhaps further system areas should be included in Category MK1 according to the decision criteria.

The MK2a category contains system areas that have an active function in secondary heat removal or provide the appropriate conditions. This classification is sensible in our opinion. Based on the present state of planning, we consider the classification of systems given in the list of systems for the MK2a as accurate.

Category MK2b contains the systems and components that reduce the radioactivity release to values below the limits specified for the designed operation. We understand this criterion for the MK2b category to the effect that in the case of assumed loss of integrity of these systems and components radiation exposure may occur in the environment, which could range from values determined for the designed operation according to § 28 Part 3 StrlSchV up to below the dose maximum values. In our opinion it is not valid and also does not comply with the state-of-the-art technology to manufacture activity-conducting components with a single quality level with as wide a bandwidth of potential radiation exposures in case of breakdown. The IAEA Safety Guides No. 50-SG-D1 /L 21/ propose the specification of potential radiation exposure limits as decision criterion. According to that, higher quality requirements are to be specified for those systems and components, which are expected to cause a higher level of radiation exposure in case of a breakdown. Alternatively, appropriate measures may be used, *e.g.*, diversity and redundancy of retention barriers and filter facilities, to justify a lower quality level. Since radiation exposure was not defined as a decisive criterion in the corpus of legislation, other projects used the alternative way.

In the MK2b classification, the applicant assumed small doses due to breakdown for the HTR module power plant facilities, which are generally small compared to the maximum doses defined in§ 28 Part 3 StrlSchV. If these values are used as the basis of determination, the chosen classification into the Mk2b category is appropriate.

Should the more detailed construction planning result in potentially considerably higher radiation exposure due to a breakdown of components in the MK2b system area, the Mk2b category has to be further divided since the above mentioned alternative route of using diversity and redundancy of the retention barriers and filter facilities is not provided for in the HTR module concept. In this case, increased requirements have to be defined for quality control measures for those system areas with higher release potentials. We do not expect this case based on the design plan on hand, however, we cannot conclusively confirm the classification of the systems from the system list to the Mk2b category at this moment. Final classification must be specified in the detailed construction plan.

To summarize, we conclude that the decision criteria for classification into the nuclear-technological categories are selected appropriately considering the functional requirements placed on the system for the HTR-Module-Power plant facility. A more precise specification may be required in relation to the Mk2b category if, as a result of the detailed construction planning, a sub-division of this category becomes necessary. The assignment of the systems and system areas to the nuclear-technological categories as described in the Safety Report or in the supplementary documentation /U 2.3–1/ is accurate and generally exhaustive. Since the applicant undertook classification only from the perspective of integrity, the final provision must provide functional specifications. For the above-mentioned reasons it only possible to make a final statement within the framework of the detailed construction plan.

According to the Safety Report, the NNK Class should generally include all those systems that were not classified into the MK1 or MK2 nuclear-technological criteria. This category also includes those system areas that have to be designed towards the protection of safety-technologically important functions regarding external impacts on integrity or stability as well as internal impacts. Appropriate evidence of this design is required for the approval procedure.

While compliance with these safety-technological requirements must be demonstrated in the MK1 and MK2a components, this is not required for the NNK components. This also applies to the MK2b system areas, which are not provided for as protected against external impacts or EVI for these or other reasons. Therefore we consider it a necessity to require an appropriate demonstration of the relevance of these system areas and their respective categories or to transfer them to category MK2a.

It is demonstrated in the supplementary documentation /U 2.3–1/ that the component auxiliary systems that are not included in the nuclear-technological classification directly should be assigned to the NNK Class in general. This is acceptable only in the case, where the operation of the auxiliary and supply facilities is not needed for the function of the nuclear-technologically classified systems. This aspect must be discussed in the detailed construction planning.

2.3.2 Ventilation Facilities

According to the specifications of the Safety Report, the applicant provides for two safety categories for the ventilation facilities.

The ML safety category contains two component groups:

- Ventilation components that minimize radioactive contamination of the environment during or after a breakdown and conduct activities on their own during such an event,
- Ventilation components that ensure the removal of heat from the rooms, in which the facilities or components needed for breakdown control are installed.

The ML safety category contains the following systems:

- Elimination of building load,
- Protected facilities with negative pressure control,
- Protected air-conditioning facilities for the control, computer, and electronics rooms, including the protected water cooling systems,
- Protected air-conditioning facilities for emergency control points.

All ventilation components that are not included in the ML category, are assigned to the category "Non-nuclear-technologically classified" (NNK)".

> The applicant carried out the classification of the ventilation facilities solely based on the requirements placed upon these facility functions. Accordingly, all the systems that are needed for breakdown control and minimization of radiation exposure after a breakdown are categorized in the higher level of classification.

We are of the opinion that the requirement of the systems' functioning during or after a breakdown substantiates an important criterion of the ventilation facility classification. What's more, in our opinion, an additional criterion for the Ml classification is the aspect of conducting activity and retaining activity according to the design of the operation.

According to the applicant's classification, the air ventilation filtering facility falls into the "NNK" quality category. This facility has the task of reducing eventual activity releases occurring during repair or maintenance procedures and makes it possible to specify a corresponding activity inventory. In addition, the air ventilation filtering facilities provide an alternative filtering possibility to ensure the protected under-pressure control for the retention of aerosols after a primary coolant leak has occurred. Under these conditions, we consider it necessary to include the air ventilation filtering facilities in the ML nuclear-technological category.

The classification of the ventilation facility must also take into account the fact that some ventilation facilities are not needed for breakdown control and other facility parts that are required for breakdown control may not be impaired in their function in case of demand. Within the present planning state of the HTR-Module, the classification of the system based on this aspect cannot been performed because the spatial classification of the individual ventilation facilities must be known. The classification of the ventilation facilities must also be completed in the detailed construction plan according to this aspect.

2.3.3 Lifting machinery and cranes

According the Safety Report supported by KTA 3902 /L 67/, the lifting machinery is classified so that the safety-technological functions are attained as they would be attained by analogous methods. They may be included in both nuclear safety-technological categories – MH1 and MH2 – as well as in the NNK non-nuclear category.

The nuclear safety categories are defined as follows:

MH1 Safety Category

Lifting machinery and cranes will be classified in the MH1 class, if the following events may result from a lifting machine failure during the transport of a load:

- Exceeding the dose maximum value as given in § 28.3 StrlSchV,
- Occurrence of a leak that cannot be sealed off in the primary gas connection, even if the dose maximum values according to § 28.3 StrlSchV are not exceeded.

According to the Safety Report this applies only to the reactor body crane and its handling parts. We understand the load handling parts to be:

- RDB-cover of the transport equipment,
- Crossbeams for transport and storage containers,
- Locking cross-beam,
- Assembly picots, break picots in concrete locks,
- Lifting mechanisms and load handling facilities for the site.

MH2 Safety Category

The lifting machinery and cranes are included in the MH2 category if an immediate risk of activity release caused by a breakdown of the lifting machinery during load transport is anticipated.

The present design, according to the applicant's statement, does not plan for hoisting devices and cranes of the MH2 class.

All hoisting devices and cranes, which are not categorized into the MH1 or MH2 class, are not to be classified according to nuclear technology (NNK).

> We consider the division into three categories for the hoisting devices and cranes sensible and accurate, especially since the KTA regulation 3902 is also based on two higher-value categories (additional and increased requirements) and a class corresponding to the conventional body of regulations. On basis of the mentioned criteria of classification we consider it correct to apply the KTA regulation 3902, Section 4.3 (enhanced requirements) to hoisting devices of the MH1 class and the KTA regulation 3902, Section 4.2 (additional requirements) to hoisting devices of the MH2 category.
>
> We consider the classification of the reactor building crane and the associated load lifting devices into the MH1 class corresponding to KTA regulation 3902, Section 4.3 as correct.
>
> We reconfirm the statement of the applicant that all other hoisting devices of the NNK class mentioned in the applicant's statements should be classified based on possible releasing of the loads to be transported. On the basis of the present planning status it cannot still be conclusively confirmed that due to the falling of loads on activity conducting devices, no type of radioactive substances are released, which would necessitate classification into the MH2 class. Therefore the definitive classification of the hoisting devices can be done only within the framework of the detailed construction planning.

2.3.4 Structural Steel Elements

According to the Safety Report, classification of structural steel elements covers the following system and components parts:

- Steel scaffolds, steel stairsm and steel landings,
- Pipe mountings (*e.g.*, fixed point constructions, guides, slide bearings, spring suspensions, fixed suspensions, bumpers and shock absorbers) as well as fixture mountings,
- Component supports (*e.g.*, pump frames, fixed architraves, claws),
- Safety and special constructions (*e.g.*, shock absorbers, manipulator guide rails),
- Anchoring parts.

The applicant envisions two safety categories for structural steel elements, two for nuclear technological and one for non-nuclear technology.

MS1 Safety Category

Here are classified those structural steel elements whose failure might

- Lead to exceeding the dose maximum values according to § 28 sec. 3 StrlSchV,
- Cause a primary leak which cannot be sealed off – irrespective of the exposure to radiation,
- Cause a simultaneous failure of all the lines of the intermediate cooling systems in the primary cell.

MS2 Safety Category

Here are classified those structural steel elements whose failure might

- Impair the functional performance of parts belonging to the earthquake category I,
- Amount to the loss of integrity of MK2-components in the reactor building.

We consider the exemplary listing of the typical structural steel elements to be appropriate for showing the range of applicability of the safety categories of steel structure in the current state of planning. In case of the construction planning, a detailed survey of all required structural steel elements must include, among other things, the cable-suspended constructions not mentioned in the listing.

The classification has to consider that on one hand a significance of the structural steel element itself can exist from a safety point of view (*e.g.*, in case of bumpers) and on the other hand, a dependency on the classification of the mounted components exists.

In addition, the significance concerning safety regulations can arise from the fact that structural steel elements that by themselves have no safety-related function may, through their failure, affect the functioning or the stability of a device, which is important from a safety point of view.

The classification criteria for the MS1 category adequately consider all aspects relevant for the classification of structural steel elements.

The applicant intends to classify all structural steel elements within the primary cell into the MS1 category. By this, the safety specifications mentioned in the classification criteria can be complied with. All components of MK1-classified systems are thus kept inside the primary cell of MS1-classified support structures. In addition, the surface cooler and all components, which might damage it in a redundancy-spanning way, are equipped with MS1-classified mountings.

In addition to the illustration in Table 2.4.1/2 of the Safety Report, among other things, all structural steel elements outside the primary cell that are employed for support, mounting and anchoring of MK1-classified systems, must be classified into the MS1 category.

In the MS2 category, those structural steel elements are classified whose failure may affect the functional performance of the parts belonging to the Earthquake Category I or may cause a loss of integrity of MK2-components in the reactor building. For the classification criteria for the MS2 category, we consider it necessary to classify the mountings and supports of systems that are classified into the category MK2 also into safety MS2 category if they are used outside the reactor building since the functional performance of these systems may be equally affected both inside as well as outside the reactor building. This is also applicable to all structural steel elements that can affect MK2a-classified systems in their functional performance.

We do not have any objection to the classification of those structural steel elements that do not fall under the classification criteria of the nuclear safety MS1 category or MS2, into non-nuclear category NNK.

2.3.5 Assessment Summary

The applicant has presented the classification of the

- Pressurized and activity conducting systems,
- Ventilation facilities,
- Hoisting devices and cranes, and
- Structural steel elements

in Section 2.4.1 of the Safety Report and in the supplementary document /U 2.3–1/.

The classification criteria forming the basis for the nuclear safety categories have been selected quite accurately taking into account the fact that we consider it necessary to carry out the nuclear classification of pressurized and activity conducting systems not only with regard to the integrity of the enclosure of these systems, but also from the point of view of functionality. As per the reasons mentioned above, it may be necessary to form additional categories or subgroups and to differentiate them from the remaining categories through additional classification criteria, which take into account the respective circumstances.

Inasmuch as possible, the classification of systems and components into the different safety categories has been performed quite appropriately based on the current planning status. We have mentioned and justified the changes required from our point of view (*e.g.*, exhaust air filtering facility).

Because of the reasons mentioned in the respective sections of this chapter, the finalization of the safety categories and the classification of the systems and components can only be performed in the framework of the detailed construction planning.

2.4 Building Layouts

2.4.1 Principles of the Safety Regulations Evaluation of the Building Layouts

In this chapter we are evaluating whether the layout design on hand meets the structural engineering approval provisions with reference to the precautionary measures against damage required by AtG § 7(2), 3 /L 1/. This is the case if the general requirements as defined in the laws governing public buildings with respect to structural installations as well as the safety regulations as defined in the laws governing nuclear energy with respect to structures of nuclear installations are fulfilled. The basic requirements are laid down in the respective regional building regulations (*e.g.*, NBauO /L 3/). The special safety regulations result from appropriately applying the Safety Criteria of the BMI /L 6/, the RSK Guidelines /L 10/, and the Failure Guidelines of the BMI /L 11/.

The approval procedure according to laws governing public buildings ensure that the general requirements of the regional building regulations are fulfilled by following generally recognized technological standards. The following examines further, whether the structural engineering tasks derived from the safety regulations of nuclear installations, such as

- Ensuring the stability and serviceability in accordance with the design layout,
- Protective function in case of occurrence of failures within the installation,
- Protective function in case of external impacts,
- Safe inclusion of radioactive materials and protection against inadmissible exposure to radiation or contamination

can be fulfilled through appropriate design, partitioning, layout, and construction of buildings within the framework of the basic building plan.

Special safety requirements applicable for nuclear installations are to be applied only to safety-related structures. In contrast, no additional requirements that go beyond the usual requirements of structural installations are applicable to conventionally used buildings. The decision whether a structure is safety-related or not, is based on the principles of safety precautions defined in the BMI Safety Criteria /L 6/ as per the following definition:
The safety-technological importance of the structures of the HTR module facility is determined by the fact, whether they are required directly (*e.g.*, structural protection, shielding, barriers) or indirectly (*e.g.*, load bearing and load transfer of safety regulated systems and system components) or whether they contribute, during operation in accordance with the regulations, in case of occurrence of failures of internal installation and in case of external impacts,

- To shut down the reactor safely and to keep in shutdown condition,
- To transfer the residual heat,
- To prevent an inadmissible release of radioactive materials and
- To ensure the safe inclusion as well as an adequate shielding of the radioactive inventory.

In this sense the following structures are safety-related:

– Reactor building	(UJA)
– Reactor building annex	(UJH)
– Reactor auxiliary service building *)	(UKA)
– Switchgear and emergency supply building	(UBR)
– Secured wet chilling chambers, and	(URB)
– Cable conduits from UBR to UJA	

*) limited (compare Section 2.4.4)

Structures not mentioned explicitly are only required for operational purposes of the installation.

For the purpose of assessment, in addition to the super ordinate BMI Criteria and RSK Guidelines /L 6/, /L 10/, /L 11/, and the KTA-regulations 2201.1 /L 41/ and 2501 /L 55/, the BMI Guideline /L 17/ as well as the appropriate structural engineering regulations, guidelines and standards, especially the DIN standards and the IfBt Guideline /L 26/ are taken into consideration.

2.4.2 General Layout

A potential general layout of the structures of the HTR-module installation is described in the /U 1/ Safety Report and demonstrated in the site plan. The essential structures are:

– Reactor building	(UJA)
– Reactor building annex	(UJH)
– Reactor auxiliary service building with exhaust air chimney	(UKA, UKH)
– Switchgear and emergency supply building	(UBR)
– Power house	(UMA)
– Cooling towers and pump structures	(URA, URB and URD, URE)

as well as the bridge structures and cable conduits connecting them.

As additional structures, among other things, the following are provided:

- Fuel element transitional storage (UFC)
- Gas supply center (UTG)
- Office building and staff building (UYA)

The reactor building with both the HTR modules and the annexes is situated at the center of the installation.

The reactor auxiliary service building is connected directly to the reactor building. Both buildings are separated from each other through a settlement joint and attached to each other by the staff entrance at +7.00 m and the material gate at ± 0.00 m. The exhaust air chimney is located on the reactor auxiliary service building.

The control and emergency supply building is located separately on the front side of the reactor building. It is connected to the remaining buildings through cable conduits.

The powerhouse is freestanding structure, situated in front of the reactor building. The turbine shafts of both turbo sets are oriented towards the reactor building. The cooling tower and cooling pump structures are situated adjacent to the powerhouse.

The facility site is approached through an entrance at the guarded gate. The main staff entrance is through the office and staff building, which is directly connected to the control and emergency supply building. The main entrance to the security area in the reactor building and reactor auxiliary services building is in the reactor auxiliary services building at the +7.0 m level. Material transport takes place through the truck entrance of the reactor auxiliary services building and then through the material gate of the reactor building at the +0.00 m level.

The spatial arrangement of the individual structures ensures easy access to the buildings. The entrances for staff and material are separate. The activity potential is concentrated within closed building complexes.

The reciprocal interference of the buildings during operating condition is prevented through their spatial separation or restricted by the structural separation (settlement joint between UJA and UKA) and does not require any conceptual decisions.

The turbo sets in the power house are erected in a way (turbine shafts directed towards the reactor building) that in case of turbine failure with flying fragments all safety-technological relevant structures – up to the protected chilling chambers – are located outside the probable direction of flying fragments of the turbo sets. This design corresponds to the specifications of the RSK Guideline 17.1 /L 10/.

We have no objection to the location of the protected chilling chambers in the area at risk by a turbine failure. Firstly, the probability of damage is very low and secondly, even in case of a failure of the protected cooling water supply, the surface cooler, which conducts the residual heat, can be externally fed by fire hydrants.
Similarly, the configuration of tanks with high energy content, such as the feed water tank in the power house and the clean gas storage tank in the gas supply facility, ensures that in case of an assumed complete break-away of a pressure tank no safety-related buildings are located in the main flying direction of big fragments of the tank.

Due to external impacts in the event of "shock waves resulting from chemical explosions" no effects amplifying the shock wave action are to be expected on account of the building design, as is possible, *e.g.*, in the case of a focused layout of buildings, long alleys or court yards. The distances between the buildings in the representative design are that considerable that in the event of an "earthquake" the to-be-protected buildings are not affected by the debris of structures that are not designed for such load type. Therefore, the distances between the power house and the reactor building as well as between the exhaust air chimney and the control and emergency supply building are greater than the height of the potentially collapsing structures. The reactor auxiliary service building is demonstrated for stability in the event of an earthquake in order to prevent inadmissible effects on the adjacent reactor building.

Overall, the potential general layout of the structures on hand thus conceptually complies with the safety regulations. In case of a specific design regarding the general layout of buildings it must be ensured that all the possibilities of reciprocal interference factors indicated here are considered.

2.4.3 Reactor Building (UJA) and Reactor Annex (UJH), Safety Enclosure

Reactor Building and Reactor Building Annex

The primary objective of the reactor building is to accommodate the technical systems and components of the facility and to accommodate the resulting loads during operation, in case of internal failures and to safely reduce induced shocks due to external impacts.

As additional objectives, the building plays a protective role for the technical installations in case of internal failures and in case of external impacts. Moreover, together with other components of the installation it serves as part of the safety enclosure for minimizing the exposure to radiation in the environment in case of failures.

The reactor building contains essentially the two nuclear steam generation systems consisting of the reactor pressure tank, the steam generator and the connecting pressure tank as well as important auxiliary and secondary systems. The emergency control station is accommodated in a specially protected area. The reactor-building annex is located outside the protection enclosure of the reactor building. In it are housed components of the start-up circuits and shutdown circuits, the intermediate cooling systems and the steam generator discharge.

The reactor building is divided into two modules through a central supply section The primary cell consisting of reactor-and-steam generator chamber, whose concrete walls enclose the reactor pressure tank and the steam generator, is situated within every module. The primary cells are surrounded by rooms with varying functions. The delivery room for the fuel element is situated below the reactor cell. The entire space above the primary cells is taken up by the reactor hall along with the assembly hall and the reactor-building crane.

The main dimensions are as follows:

– Length	approximately	46 m
– Width	approximately	36 m
– Height	approximately	39 m
– Foundation depth	approximately	-15.5 m

The construction is a massive reinforced concrete structure with a flat roof, with a foundation on a continuously reinforced concrete base plate. The internal structure of the building is separated from the external reinforced concrete enclosure by 30 cm width settlement joints to prevent direct action of forces due to external impacts. With regard to the reactor building crane, the bridge bears on one side of the internal structure of the building and on the other hand on the console plate of the external reinforced concrete shell. Protection against pressing groundwater from underneath is provided by a shared pressure water-holding seal for the reactor building and the reactor auxiliary services building. Both buildings are separated from each other through a joint.

The building dimensions are based on the technical building specifications for operational loads during the intended operation as well as from the point of view of shielding /U 2.4–2/. To control failures that may occur within the internal installations, the structures of the reactor building are designed against local and global effects resulting from the postulated failure of pressurized components. to this end, the following effects are assumed:

- Radiation forces,
- Reaction forces,
- Impact loads from the separated fragments of components,
- Differential pressures,
- Temperature increases, and
- Flooding.

The differential pressures and temperatures are defined for the assumption that a primary coolant pipeline (DN 65, Failure Pressure Discharge) or a live steam line bursts and the pressure relief of building occurs through the chimney into the atmosphere.

The design pressure for the primary cells and the external walls is 0.3 bar. The design temperatures for the concrete construction of the primary cell are 90° C during normal operation and 150° C during anomalous operation or in case of failures /U 2.4–1/. Other internal facility failures that of importance for the safety-related evaluation of the structure, are discussed in the following precautionary measures:

It is ensured that fragments resulting from the failure of rotating component parts do not penetrate the enclosure surrounding them. Collapse of heavy loads is ruled out because of the enhanced design requirements of the reactor-building crane.

Structural protective measures, such as firewalls and fire passages, have been implemented to prevent fire hazards.

The structural protection concept against external impacts entails the design of the reactor building for the following events:

- Earthquake,
- Aircraft crash, and
- Shock waves from chemical explosions.

The reactor-building annex is designed to withstand earthquakes only (see section 5.6).

In addition following impacts are considered:

- Fire in the surrounding area,
- Lightning strike,
- Wind, storm,
- Snow, rain, hail,
- Flood, low water level,
- Hazardous gases.

The design for earthquake is carried out according to the KTA regulation 2201.1. Burst shock waves from the possible failure of tanks with high-energy content are considered as impacts. Impacts due to aircraft crash and shock waves from chemical reactions are based on the design loads as per RSK guideline /L 10/ or BMI guideline/L 17/. The dimensions of external walls and the roof of the reactor building are such that complete protection is achieved against aircraft crash. All safety-technologically relevant parts are designed against the induced shocks from external impacts.

The structural engineering analyses are carried out for the combinations of loads due to normal operation and unusual loads. A temporal superimposition of simultaneously occurring loads is carried out taking the sequential processes into consideration /U 2.4–2/.

> The construction of the reactor building and of the reactor-building annex with reinforced concrete corresponds to the state-of-the-art science and technology for structures of this type and usage.
>
> The basic requirements expected of the building for the intended operation according to the appropriate regional building regulations (*e.g.*, NBauO /L 3/) are ensured through the demonstration of stability and operability within the framework of the laws governing the approval procedure for buildings. With regard to the design loads, a design based on the technical building specifications represents an adequate basis for the non-installation-specific operational loads. Installation-specific operational loads, *e.g.*, load designs, must be examined based on appropriate documents before construction and then be submitted as basis for the static analyses. This testing has no impact on the design of the reactor building.

In our opinion, it can be expected that the stability of the concrete structures of the primary cells can also be demonstrated in case of the assumed concrete temperatures during the normal operation and anomalous operation or in case of failures.

Due to the spatial division, the use of concrete as building material, and the planned shielding thickness, we consider, from a conceptual point of view, the building appropriate to fulfill the requirements of the BMI Safety Criteria 2.3 and 2.4 /L 6/ or the RSK-LL 5.2(3) /L 10/ with regard to exposure to radiation in the environment and to direct radiation within the installation. No special design requirements are needed for the reactor building as part of the safety enclosure, as they are according to the BMI Safety Criteria 8.1 and 8.2 /L 6/ for gas-tight and pressure-resistant safety enclosures of light-water reactors. Due to the reliable enclosure of radioactive materials in the fuel elements and the resulting low activity of the primary coolant, the rising internal pressure of the building can be directly let out through discharge openings into the chimney even in case of the postulated breakage of a primary coolant-conveying pipeline. We have described detailed specifications of this complex in the following section "Safety Enclosure".

In order to guarantee the safety-technological functioning of the installation even in case of failures occurring in the internal installations, additional requirements of the structural design and construction of the building are specified. According to BMI Safety Criterion 8.2 /L 6/, the building structures are to be designed in a way that they withstand the static, dynamic and thermal loads during the intended operation and in case of failures.

Considering all the local and global impacts from the postulated failure of pressure-carrying components including the corresponding combinations of loads, these requirements within the scope of the RSK guideline 21 /L 10/ can also be achieved. Hence, conceptually, it is ensured that the structure fulfils its function of protecting the installation parts or of unaffected installation components in case of internal failures.

During operation in accordance with the design and in case of the assumed failures, the concrete is exposed to thermal loads. The rule for buildings is that concrete structures of load-bearing parts should not be exposed over longer time to temperatures higher than 250° C /L 159, L 160/. In case of higher temperatures, fireproof and highly refractory concretes (so-called fire-proof concrete) have to be employed. Even though concrete belongs to the non-inflammable materials, its properties change very much at higher temperatures. To some extent, the strength may initially rise slightly through intense hydration and drying, but in general, depending on the age of the concrete, type of cement and additives, the compression strength drops by approximately 10 to 20% in the temperature range up to 300°C, while it decreases very steeply from 300 to 450°C. Due to the definite reduction in strength from 300°C onwards, concrete should be exposed to working temperatures of more than 250°C only for a short time and not for longer durations. According to DIN 1045 /L 159/, the empirical values for the compression strength and the modulus of elasticity of the employed concrete need to be experimentally determined in case of constantly acting temperatures above 80°C. Up to 80°C of concrete temperature, the empirical values of the properties of concrete at room temperature are applicable.

According to the Technical Document /U 2.4–1/, temperatures of approximately 40 to 80°C (design temperature = 90° C) result for the concrete of the primary cell during normal operation so that for the strength analysis a reduction of the empirical values of the properties of concrete is not necessary. For anomalous operation and for postulated failures (*e.g.*, short time failure of the surface cooler), concrete temperatures up to 145° C (design temperature = 150° C) are to be expected, so that an evaluation and possible decrease of the characteristic values for concrete are required. According to DIN 1045 /L 159/, *e.g.*, in case of a short time temperature effect (approximately 24 h) of 150° C, the calculated value of the strength of concrete would have to be reduced by about 12% without more exact experimental analyses. From the viewpoint of the calculation it may be expected that the stability analyses of the concrete structures of the cell as well as of the reactor pressure vessel can be demonstrated for the design temperatures of concrete within the framework of the construction planning.

Whether the structural engineering fire protection measures are suitable to conform to the requirements of the BMI Safety Criterion 2.7 /L 6/ will be examined during the assessment of the fire protection design and not discussed here in depth.

The reactor building and a part of its annex have to be designed as safety-technologically critical structures for the purposes of the protection of the facility against external impacts (EVA). This requirement follows from the BMI Safety Criterion 2.6 /L 6/ and the rational application of the RSK-LL 18 and 19 /L 10/. In Section 5.6 of this report it is stated that the measures planned in the Safety Report /U 1/ for EVA are suitable for achieving the goals of protection derived from the above criterion. In the following text, aspects of building design are added to this general evaluation. We agree to the designed scope of the reactor building against all EVA-load cases and the design of the annex against earthquakes.

Aside from the indicated impacts, additional site-specific events or combinations of events (*e.g.*, damages due to mining operations, landslides, and flood waves) have to be taken into account. In our opinion, these events have no effects on the design of the building, but do have impacts on the structural construction.

The impacts of wind, storm, snow, hail, and ice formation not further treated in the safety regulations, are considered in the construction approval procedure on basis of the technical building specifications in the design.

We have given our observations regarding the design loads in case of earthquake in Section 5.6. The design loads in the event of an aircraft crash correspond to those of the RSK-LL 19 /L 10/. The intended design for complete protection of the reactor building ensures safety against a backside chipping-ff of concrete parts in addition to the protection against penetration. Thus, a direct inadmissible impact on safety regulation system parts situated behind the impact location, through fragments, is prevented.

Another further conceptually significant feature of the reactor, the structural separation between the internal and external constructions, should be highlighted. The induced vibrations due to an impact are, thus, not transferred directly but only through the foundation into the inner surfaces and walls and thus attain significantly lower values. Local coupling of the internal structure of the building with the external reinforced concrete enclosure takes place through the reactor-building crane. However, since the crane bridge is positioned horizontally only on the internal side of the building, but positioned on the external side in a sliding manner, a direct transfer of force is possible only through local frictional forces. We estimate this impact as low for the entire stress of the building.

The load event shock wave corresponds to the design load for a site-independent design according to the BMI guideline /L 17/. Site-specific characteristics may potentially demand additional analyses (compare Section 1.7).

Based on the design for all relevant EVA events and assumptions for their respective design loads and loads combinations according to the state-of-the-art of science and technology, it can be assumed that the reactor building and its annex conform to their functions with regard to external impacts.

To summarize, we consider the proposed design of the reactor building and the reactor-building annex with regard to division, design and construction as suitable for fulfilling the safety regulations during operation, in case of internal failures and in case of external impacts.

Safety Enclosure

The function of the safety enclosure of the HTR two-module power plant, according to the Safety Report and Document /U 2.4–3/, is to minimize the exposure to radiation into the environment for failures with activity release. The safety enclosure covers

- The reactor building,
- The facilities for building stress relief and for the exclusion of the ventilation,
- The protected vacuum maintenance.

The applicant's design plans for differentiated procedures according to course of events, in order to achieve the protection objective mentioned at the outset. Therefore, the applicant bases the design of the safety enclosure on two different courses of events described in the following.

In case of a primary circuit leakage from the leak section of a torn-off measuring pipeline of a maximum diameter of 10 mm, no excessive pressure over the ambient air is formed in the reactor building. The failure is detected by monitoring the ambient air activity. In case of such an event, the applicant plans to vent the reactor building by means of the Protected Vacuum Maintenance of pressure equipped with suspended particle filters and activated carbon filters. This installation is designed for a volume flow of 2.5 cubic meters/second and provided with emergency power supply. The maximum leakage of the reactor building amounts to 50 volume percent/day at 2 mbar differential pressure. A design of the protected vacuum maintenance against assumed external impacts is not planned.

In case of the assumed rupture of a non-blockable DN 65-connecting line of the primary cooling circuit, excess pressure builds up in the reactor building. In order to reduce the excess pressure, the primary coolant is led off through discharge channels to two overflow openings in the external wall of the reactor building and from there further through the chimney into the environment. The discharge channels to the exhaust air chimney are closed through valves, which open at a pressure of 1.1 bar and close again automatically after pressure equalization. Additionally the discharge canals are equipped with a lock-up valve, which is constantly open and can be closed manually in case of a pressure discharge, should the pressure discharge lock-up valves fail to close. Thus, a directed airflow can be produced after pressure equalization in the reactor building by means of the Protected Vacuum Maintenance. It is planned to operate this ventilation unit during the nuclear heating phase after the rupture of a DN 65-line, in order to minimize the activity emission into the environment. The primary cells and the external walls of the reactor building are designed for a pressure of 0.3 bar.

For the justification of his design of the safety enclosure, the applicant describes that due to the favorable activity-retaining properties of the fuel elements, the core and reactor designs and the use of Helium as coolant, no special requirements exist regarding a safety enclosure. In this connection the applicant refers especially to the results of his calculation of the radiation exposure, according to which, in case of the rupture of a DM 65-connecting line as well as in case of unfiltered discharge during the nuclear heating phase, the dose will clearly stay below the maximum values according to § 28 Section 3 of StrlSchV.

We are in agreement with the applicant that the safety-technologically favorable properties of small high-temperature reactors have to be considered in case of the design of the safety enclosure of a HTR module power plant. The safety-technological characteristic of the HTR module power plant essential for the evaluation of the safety enclosure is the low activity release from fuel elements during intended operation and during failures. Our assessment of the statements of the applicant regarding the release behavior of the fuel elements and regarding the activity release from the facility confirms the statement that for failures with activity release, even without consideration of the filters, the exposure to radiation in the area would lie below the dose maximum values according to § 28 Section 3 of StrlSchV. Thus, waving the requirement for a pressure-bearing containment is justified and the filtering of the discharges in the nuclear heating phase becomes important only with regard to the minimizing requirement of the radiation protection regulation. The design of the safety enclosure planned by the applicant is suitable for fulfilling this function.

The reactor building is designed for pressure loads during the assumed courses of events. The volume flow of the Protected Vacuum Maintenance is adequately dimensioned to maintain a lower pressure compared to the external atmosphere in case of the indicated building leakage after pressure equalization through stress relief facilities of the building. However, prerequisite for the targeted minimization of radiation exposure into the environment through directed and controlled discharge of the activity is that in case of failures with pressurization in the reactor building the pressure relief occurs reliably. Otherwise, in case of such an accident a ground level release would take place through the building leakages. In case of the presently planned design, the pressure discharge lock-up valves open in case of all failures of pressure discharge that lead to an internal pressure of the building of at least 1.1 bar. However, pressure discharge failures with medium leaks are possible that lead to the pressurization of the building below the threshold value at which the pressure discharge occurs through the lock-up valves. We consider it necessary that these cases are analyzed by the applicant within the framework of construction planning and that it is demonstrated through which measures a ground level release of radioactive substances is limited in this case.

In case of observance of this requirement we have no objection to the design of the safety enclosure for the HTR-module-power plant.

2.4.4 Reactor Auxiliary Service Building (UKA)

The reactor auxiliary service building contains auxiliary installations of the reactor installation, such as the

- Helium purification unit,
- Systems for the treatment of radioactive waste materials,
- Ventilation facilities,
- Storage facilities for new fuel elements,
- Sanitary wing,

as well as other facilities.

The main dimensions are as follows:

- Length approximately 46 m
- Width approximately 26 m
- Height approximately 17 m

The bottom-most level is situated at –9.5 m.

Overall, there are six full floors and one partial floor. The building is a reinforced concrete structure with flat roof. Some internal walls are brick-laid. The foundation is a flat foundation on a reinforced concrete foundation plate. For the protection against groundwater under pressure, the reactor auxiliary service building and the reactor building are provided with a common pressure water holding seal sump up to the top level of the ground.

The design of the building is construed for the intended operation loads as well as according to points of view pertaining to shielding measures. Differential pressures and temperatures are considered as internal failures; they occur under the assumption that a line of the Helium purification unit breaks and building relief into the surroundings takes place. Against external impacts an earthquake-proof design according to the KTA regulation 2201.1 /L 41/ is provided for water-proofing the floor and the structures that support it. The other main load-bearing structures are designed for stability according to DIN 4149 /L 73/ for a site-dependent safety earthquake.

Certain safety-technological regulations are put forth for the reactor auxiliary services building. On the one hand, the global stability is to be ensured in case of external impacts in order to avoid debris loads on the adjacent reactor building or inadmissible damages to the sealing construction. On the other hand, the building has to contribute to the prevention of radioactive fluids leakage.

The stability requirement is fulfilled through the operational design as well as through the consideration of the pressure buildup from the postulated failure of a high-energy pipeline and through the differentiated design of the building against earthquake.

The differentiated procedure in case of the earthquake design – building tray according to KTA regulation 2201.1, further building stability according to DIN 4149 – is legitimate due to the different safety-technological importance of the structural parts. The reactor auxiliary service building contains no system parts that carry out safety-related functions. Possible leaks in active components and systems do not lead to inadmissible release of the radioactive inventory. Therefore, the building can be classified as category II structure and thus as non-safety-regulated in the denotation of the KTA regulation 2201.1. However, it has to be shown that the immediately adjacent reactor building and the shared sealant construction are not affected by the effects and damages possibly resulting in the building in the case of an earthquake. This requirement is fulfilled for a safety earthquake at the site by the procedure used to demonstrate the stability of the building according to DIN 4149 for structures in nuclear plants with comparable protection designs. Here, the seismic design loads are to be assumed according to the state-of-the-art of science and technology. Thus, it is ensured that the earthquake seismic impacts do not lead to the failure of the main load-bearing construction of the reactor auxiliary services building and resulting damages remain limited. An inadmissible damage to the reactor building and the sealant construction from debris loads can thus be excluded.

The earthquake design of the building tray, consisting of the groundwater seal and the reinforced concrete structures that support them according to KTA regulation 2201.1 results less from a corresponding requirement of the reactor auxiliary service building itself. Rather this is a consequence that the sealing tray is shared with the reactor building and that therefore the design requirements for both buildings can not to be separated.

The function of the groundwater seal, as an external structural sealing, is to prevent the penetration of water into the building and in this manner to protect safety-technologically important systems and system parts against the damaging impact of moisture and water. For the reactor building, the penetration of groundwater can present a safety risk, since components needing protection are set up there. Within the reactor auxiliary services building such components are not found. The requirements of the dimensioning and construction of external structural sealing are specified, *e.g.*, in the KTA regulation 2501 /L 55/ and the DIN 18195 /L 74/ and already implemented in comparable nuclear plants.

We consider the common sealing construction of reactor building and reactor auxiliary service building in connection with the planned design of the structures serving as sealing carrier suitable for protecting the structures against groundwater penetration.

The leakage of radioactive fluids released in the rooms of the reactor auxiliary services building from the structure into the underground is prevented by means of the internal seal. The planned floor coatings in connection with room drainage are suitable for ensuring an adequate protection against the leaking of radioactive fluids. In addition, the external structural sealing forms an additional barrier against a contamination of the underground.

2.4.5 Power House (UMA)

In the power house, essentially are located the components of the water steam circuit with the two turbo sets and the components of the steam process system.

The main dimensions are as follows:

- Length approximately 63 m
- Width approximately 49 m
- Height approximately 25 m in the main wing
  approximately 19 m in the side wing

The fully cellared building is subdivided into a main wing and a lower side wing. The load-bearing structure of the power house consists of a reinforced concrete framework structure all the way up to the turbine floor. The part on it is implemented as a steel construction. The foundation is a flat foundation on a reinforced concrete-foundation plate.

The foundations for the turbo sets are implemented as reinforced concrete carrier grid, which discharge their loads through springs onto columns. The reinforced concrete foundations for the feed water pumps are also positioned on springs and are separated from the surrounding parts of the power house through joints.

The design of the building is carried out for operational loads according to the intended operation. The building is not designed against internal facility failures or for external impacts.

> The power house does not contain any systems or components whose function is safety-technologically relevant. It need not fulfill any structural engineering functions with respect to the safe enclosure of radioactive materials and the protection against inadmissible exposure to radiation. The requirements of the building are determined only with respect to the operational purposes and are covered by the planned design according to the specifications of the public building regulations.

2.4.6 Switchgear and Emergency Supply Building (UHR)

In the switchgear and emergency supply building are located switching and control technology installations, *i.e.* the watchtower, the emergency power supply, and other facilities. Cables are led from here to the other buildings through underground cable trenches.

The main dimensions are as follows:

- Length approximately 39m
- Width approximately 38 m
- Height approximately 13 m or 17 m.

The load-bearing construction of the building consists of a reinforced concrete framework and stiffening reinforced concrete wall disks and ceilings. The foundation is made as a flat foundation on a reinforced concrete-foundation plate.

The building is designed for operational loads and against earthquakes including resultant effects.

> The switchgear and emergency supply building contains systems and components, which serve to shut down the reactor and the residual heat discharge. The construction of the building has the safety-technological function to accommodate these system parts, to reduce their loads and to protect them above all against failures occurring facility-internal failures and against external impacts. The building meets these requirements on the one hand through the partitioning regarding the spatial separation of redundant electrical facilities. On the other hand, the requirements of the BMI Safety Criterion 2.6 /L 6/ for the protection of the installation against external impacts are fulfilled through the design of the building against earthquake and subsequent loads.

The design for aircraft crash and explosion shock wave is not necessary because even in case of a partly or fully destroyed switchgear and emergency supply building, the reactor is automatically switched off by means of the reactor protection system and the monitoring of the installation and because the long-term securing of sub-criticality is executed by the emergency control station in the reactor building. The residual heat release is ensured in this case through the surface cooler designed for these events, which can be fed externally through fire hydrants.

2.4.7 Intermediate Fuel Element Storage (UFC)

The burnt-off fuel elements are kept in the intermediate fuel element storage in transport and storage containers for waste disposal. The storage capacity is 8 containers and can be extended if required.

The approximately 40 m long and 16 m wide structure consists of a flat foundation plate on which reinforced concrete shielding boxes are assembled in two series by prefabricated construction (outdoor assembly). The boxes enclose the containers and are provided with a covering for weather protection. The after-heat of radioactivity is removed by natural draught cooling through ventilation openings in the boxes. The individual storage locations are approached by an overhead crane, which is positioned on a separate steel beam structure. The transportation of the containers at a lower height in the alley lying in between them is possible due to the two series arrangement of the boxes. There are no special requirements for the structural design of the intermediate fuel element storage.

The ball-shaped fuel elements used in the HTR module have the characteristics of high fission product retention. In addition, the storage of burnt-off fuel elements takes place in transport and storage containers that should fulfill through their design and construction all safety regulations such as sub-criticality, residual heat discharge, shielding and enclosure of radioactive materials as well as integrity in the case of external impacts. For this reason, no special safety-technological requirements that exceed the usual requirements for buildings are specified for the building including for the overhead crane. A capacity extension is possible.

2.4.8 Gas Supply Center (UTG)

The gas supply installation serves as the storage and supply unit of the installation with Helium and nitrogen. The pure gas-storage vessel with Helium, which is used as primary coolant, and the nitrogen bottles are positioned flatly on independent foundations in an outdoor installation. The components required for the feeding of the gases are assembled at ground level on a foundation plate in an 11 m long and 5 m wide lightweight-steel construction hangar. A hose/cable canal made from concrete mixed at the site with prefabricated covers connects the gas supply center with the reactor auxiliary services building.

Requirements of the structure come from the general structural specifications in combination with the regulations of the conventional control equipment valid for installations of this type. Further requirements from considerations of failures do not exist.

2.4.9 Cooling Tower Structures and Cooling Tower Pump Structures (URA, URB and URD, URE)

The main cooling water system, the conventional and the nuclear secondary cooling water system and intermediate cooling system as well as the secondary cooling water system and intermediate cooling system for the start-up and shutdown circuits release their heat to the atmosphere via the hybrid cooling tower. The protected secondary cooling water system and intermediate cooling system are cooled through two separate wet chilling chambers. The pump structures are directly associated with to or built below the coolers.

The hybrid-cooling tower consists of four cells and connects the wet and dry heat exchange with one another. Depending on weather, the formation of cooler mist is avoided through the combined operation of both systems. The heat exchanger elements of the dry part, which consist of tubes bundles made from polyethylene, are arranged in the upper section of the cooling tower. In the lower half the wet part is connected with conventional trickling grids made from polypropylene. The load-bearing structure of the cooling tower is constructed in reinforced concrete. Housing, basin and other parts are constructed from concrete mixed at the site or as prefabricated products.

No special design requirements are issued for the hybrid-cooling tower. In contrast, both wet chilling chambers and their associated pump structures are designed for earthquakes. The cooling water systems connecting to the hybrid-cooling tower have no safety-technological function. Accordingly, no special design requirements that exceed the usual structure specifications are made for the structure.

The protected intermediate cooling water system and secondary cooling water system, which is re-cooled by the wet chilling chambers, must carry away the heat taken up at the cooling locations in the primary cells during operation and during failures. The system is built in a double-stranded fashion. The wet chilling chambers and their pump structures conform to the safety-technological importance of the system through their separate construction and through the earthquake-resistant structural design.

2.4.10 Exhaust Air Chimney (UKH)

The exhaust air chimney serves to discharge the exhaust air from the ventilation facilities of the reactor building and reactor auxiliary services building. In addition, the pressure relief of both buildings takes place through the chimney. The exhaust air chimney is anchored on the roof construction of the reactor auxiliary services building at approximately +18.20 m. The height of its opening lies at +60.00 m. A support stay is provided at approximately +38.00 m.

The chimney consists of a steel plate construction with an internal diameter of approximately 2.60 m. It is not designed as earthquake-resistant.

The exhaust air chimney is of importance with reference to the radiological contamination of the environment through the emission of radioactive materials and the impairment of safety-technologically critical structures in case of external impacts. We have no objections against the planned height of the opening of the chimney from radiological point of view, without consideration of location-dependent characteristics.

Since the exhaust air chimney is not designed as earthquake-resistant, as consequence of this event through a possible collapse of chimney safety-technologically critical structures, may be hit by debris. In this context, the reactor building and – to some extent – the reactor auxiliary service building is affected. The distance of the switchgear and emergency supply building to the chimney site is so large that, in case of the exemplarily presented design, debris loads for this building need not be assumed. The reactor building is protected against aircraft crash and thus by default also against the lower loads from chimney debris.

The adjacent reactor auxiliary service building is itself demonstrated for stability against debris loads due to earthquake type load to prevent significant damage of the reactor building and the shared sealing construction. However, it cannot be ruled out that falling debris from the exhaust air chimney and damages to its anchoring on the roof to the reactor auxiliary service building cause damages that endanger the stability of the building in some areas. For this reason, we consider it necessary that the stability in case of the earthquake load type is also demonstrated for the exhaust air chimney or that the effects of debris loads from a possible collapse of the chimney onto the reactor auxiliary service building is considered and are demonstrated as acceptable.

2.4.11 Canals and Routings for Cables and Pipelines in the Ground

Between the individual structures there are connecting structures in the form of trenches, bridges and conduits for pipelines and cables.

The cable conduits are made of reinforced concrete and are laid in the ground. The pipe bridges are made of composite steel structures. Cables and pipelines are freely laid in the ground for additional routings. The cable trenches between the switchgear and emergency supply building (UBR) and the reactor building (UJA) are designed as safety-technologically critical connections and as earthquake-resistant.

> Special requirements regarding the design, construction and guiding of the connecting structures originate from the respective need for appropriate protection against external impacts and the reciprocal protection of the individual redundancies. The requirements of protection against external impacts are fulfilled through the earthquake-resistant design of the cable conduits between UBR and UJA. The protection of the individual redundancies is ensured through the spatial and constructive separation of the cable conduits as well as through the planned gravelling in fire passages.

2.4.12 Other Building Facilities

Other building facilities that are listed in the Safety Report are buildings that are only required for operational purposes of the facility as well as the office and staff building and the gatekeeper building. These structures have no safety-technological specifications. The requirements of the structures are covered by the general building specifications.

2.5 Reactor Core

2.5.1 Overview

The reactor core consists of approximately 360,000 spherical-shaped fuel elements, which are arranged within the graphitic and metallic core mounting components in the reactor pressure vessel. Apart from these components we also treat in this chapter the shutdown devices also located in the reactor core vessel and the physical and thermodynamic design of the reactor core in the reactor pressure vessel.

The data of the equilibrium core characterizing the equilibrium reactor core are summarized in Table 2.5.1–1. Further data concerning the primary core and the run-in phase (transfer from the primary core to the equilibrium core) are indicated in technical documents /U 1, U 2.5–1.1/. As per them, the primary core consists of 50±5% fuel elements, 50±5% moderator elements and absorber elements, in which not more than 2.0+0.5% have a latent reactivity.

The fuel elements used in the primary core require the use of 7 grams of Uranium heavy metal identical to the fuel elements intended for the equilibrium core, but with a reduced U 235-enrichment of 4.2+0.8 percent by weight. In order to limit the fuel element or the particle performance to the range of 4 kW/BE or 250 mW/particle safeguarded by radiation experiments, the fuel element re-charging takes place during the first 70–100 full-load days still with the primary core-fuel element. After 80–100 full-load days, a transfer to a different fuel element type, also with 7 g Uranium and 6 ± 1 percent by weight U 235-enrichment takes place.

The refueling strategy (exchange of moderator elements and absorber elements for fuel elements) for the run-in phase is established in a way that the admissible fuel element and particle performance can be observed and the fine-tuning regarding the reactivity can be carried out with the help of the variable core circulation rate as well as the provided reactivity rate action in the control and shutdown system /U 2.5.1-1/.

The design of the reactor core was performed under consequent use of the possibilities given in the case of small high temperature reactors in order to attain favorable inherent safety features. These are particularly the following:

- The use of fuel elements, which release only very low quantities of fission products up to the design temperature of 1620°C,
- The design–specified limit of the maximum possible fuel element temperatures to values below the fuel element-design temperature of 1620°C in case of failures,
- The residual heat discharge from the core to the surface cooler through passive heat transport mechanisms,
- The shutdown through the reflector rod system through abdication of shutdown rods that are to be inserted into the fuel element charge.

These properties are essentially achieved through

- The use of fuel elements with low enriched oxide fuel in case of a Uranium charging of only 7 g Uranium per fuel element,
- The choice of a slender core geometry with a diameter of only 3.0 m at a height of 9.4 meters for the pebble bed,
- The selected low average power density in the reactor core and the uniform power density distribution achieved through the multiple pass of the fuel elements,

- The wide temperature span of 750°C between the BE operating temperature and the design temperature of the fuel elements based on the interaction with the negative temperature coefficients of reactivity, and
- The restriction to a load cycle range of 50% to 100% of the nominal capacity.

According to the BMI safety criteria for nuclear power plants, a nuclear power plant must be designed and operated in a way that the reactor installation can be shut down safely any time during the intended operation and during failures and kept in the shutdown condition, so that the residual heat can be discharged and the exposure of the staff and the environment to radiation is also kept as low as possible even below the maximum values for doses under observance of the state-of-the-art of science and technology.

The planned design features of the reactor core of the HTR-module are, according to our opinion, suitable especially for achieving the higher-level safety objectives indicated in the BMI Safety Criteria. In the following sections of this chapter we deal with this in connection with the design of

- The fuel elements, moderator elements and absorber elements,
- The metallic and ceramic core components, and
- The shutdown devices

as well as in case of the evaluation of the physical and thermodynamic design of the reactor.

With regard to the construction and the operation of the reactor core we have evaluated whether the specifications of the applicant are adequate enough in order

- To determine the physical reactor characteristics with sufficient accuracy and
- To demonstrate the observance of design values of temperature, pressure and neutron fluence of the reactor pressure vessel (RDB) and the RDB-components.

We establish that the specifications for the setup of the reactor core, the bandwidths of the core composition and the Uranium 235-enrichment of the BE are sufficient to determine the indicated characteristic values and design values. Together with the design data of parts adjacent to the reactor core, *e.g.*, control- and shutdown elements, blower, steam generator and surface cooler, as well as through operating values and set values for the reactor, evaluations regarding the observance of the paramount safety objectives can be carried out.

The globally demonstrated refueling strategy during the run-in phase of the reactor core /U 2.5.1–1/ takes the preceding safety objectives into complete consideration:

- Limiting of the performance per BE,
- Limiting of the particle performance,
- Limiting of the surplus reactivity

We are of the opinion that, based on the variation possibilities of the core composition, the circulation rate and the U 235-enrichment of the BE a refueling strategy can be established in detail – possibly, with additional operational restrictions – which follows the indicated established safety objectives.

Table 2.5.1–1: Nominal data of the equilibrium core

Thermal performance	200 MJ/s
Core Diameter	3 m
Average core level	9.4 m
Average power density	3.0 MW/cubic meters
Average primary cooling medium temperatures	250/700° C
Primary circuit pressure	60 bar
Number of fuel elements	360,000
Number of rad. enrichment- zones	1
Number of fuel element passes	15
Reactivity rate action for load cycle	1.2%
U 235-enrichment	8.0 ± 0.5 w/o
Heavy metal charging	7 g/BE
Average fuel element dwell period	1,007 VLT*
Average fuel element pass time per pass	67 VLT*
Target burning off	80,000 MWd/Mg U
Conversion rate	0.47
Total power density form factor**	1.8
Neutron loss from the discharge	13.7%
Maximum neutron fluence at the side reflector*** (E higher than/equal to 0.1 MeV)	1.8×10^{22} cm^{-2}
Fuel stock of the core	
– Heavy metal (without fission products)	2,396 kg
– Fissionable material	107 kg

* VLT: full-load days

** averaged over all fuel elements at site

*** after 32 VLa (full-load years)

2.5.2 Fuel Elements, Moderator Elements, Absorber Elements

2.5.2.1 Fuel Elements

Construction

According to the details in the Safety Report /U 1/ and in the technical document /U 2.5.2–1/ the applicant plans to employ the fuel elements described below.

The fuel elements are spheres of 60 mm external diameter. An inner spherical zone of 50 mm diameter contains the fuel in the form of spherical UO_2-cores (0.5 mm ø), which are surrounded by several pyrolytically separated carbon layers and a silicon carbide (SiC) layer. These coated fuel particles (TRISO particles) are embedded in a carbon-matrix and uniformly distributed. The matrix, which is based on natural graphite and electro-graphite is – together with the coated fuel particles – quasi-isostatically pressed with a synthetic plastic resin-bonding agent under high pressure without additional heating.

An external fuel-free shell of approximately 5 mm thickness made of the same matrix material is pressed onto the inner fuel-containing zone. After the turning to final size and under consideration of dimensional changes during the subsequent heat treatment steps, the coking of the bonding agent takes place under inert gas and up to 800°C as well as the final high temperature treatment up to 1950°C under vacuum for the degasification and purification of the fuel elements. The design data of the fuel elements are indicated in Table 2.5.2–1.

Table 2.5.2-1 Design data of the fuel element (equilibrium core)

Fuel cores of the TRISO-particle

Fuel composition	UO_x $x \leq 2.01$
U 235 enrichment	$8.0 \pm 0.5\%$
Diameter	500 ± 20 µm
Density	≥ 10.4 g cm^{-3}

Coating of the TRISO-particle

Buffer layer, thickness	95 ± 18 µm
Buffer layer, density	≤ 1.05 g cm^{-3}
Inner PyC-layer, thickness	40 ± 10 µm
Inner PyC-layer, density	1.9 ± 0.1 g cm^{-3}
Inner PyC-layer, anisotropy (BAF)	≤ 1.10
SiC-layer, thickness	35 ± 4 µm
SiC-layer, density	≥ 3.18 g cm^{-3}
External PyC-layer, thickness	40 ± 10 µm
External PyC-layer, density	1.9 ± 0.1 g cm^{-3}
External PyC-layer, anisotropy	≤ 1.10

Matrix/fuel element

Fuel element weight	204 g
Matrix material	A3
Matrix density	1.75 ± 0.02 g cm^{-3}
External diameter	60 mm
Thickness of the fuel-free shell	5 ± 1 mm
Thermal conductivity 1000°C	≥ 25 $Wm^{-1}K^{-1}$
Standard corrosion 1000°C	≤ 1.3 mg $cm^{-2}h^{-1}$
Ultimate strength	≥ 18 kN
Anisotropy thermal elongation	≤ 1.3
Drop strength (4 m/spherical bed)	≥ 50
Wear (revolving drum 100 h)	≤ 6 mg h^{-1}
Heavy metal content per BE	7.09 g
Number of coated (small) particles per BE	11,600
Share of free Uranium	$\leq 6 \times 10^{-5}$
Impurities, boron equivalent	≤ 1.3 ppm B

The individual components of the fuel element have to fulfill the following functions:

– Fuel core of the TRISO-particle

In the fuel core the thermal power is produced through fission of the nuclear fuel. Here the bulk of the fission products are generated. In order to achieve a uniform charging of the fuel particle to the greatest possible extent, *i.e.* low voltage peaks in the coating, a spherical core form is targeted.

– Buffer layer

The inner buffer layer consists of relatively porous pyrocarbon (PyC). Its function is mostly to make available voids to restrict the internal pressure caused by the gaseous fission products resulting during nuclear fission. In addition, the buffer layer with its low strength produces a mechanical decoupling between the fuel core void, which swells up with increased burn-up, and the high-density layers, which are the determining factors for guaranteeing the fission product retention. Furthermore, the kinetic energy of the fission products from fissions in the periphery of the fuel core is reduced in the buffer layer and thus protects the high-density layers against damages.

– Inner high-density Pyrocarbon-layer

The inner high-density isotropic Pyrocarbon layer (ILTI: inner low temperature isotropic) forms an initial pressure barrier against the inner fission gas pressure and thus reduces the internal pressure loading of the following SiC-layer, which allows only limited tensile stresses. Furthermore, the high-density PyC-layers almost fully retain the fission gases and the similarly reacting iodine under normal operating conditions.

– Silicon carbide-layer

The function of the Silicon carbide-(SiC-)layer is to retain all fission products, especially in case of the temperatures assumed during failures of the HTR-module.

– External high-density Pyrocarbon-layer

Above all, the external high-density isotropic Pyrocarbon-layer (OLTI: outer low temperature isotropic) serves as mechanical protection for the SiC-layer in case of handling of the finished coated fuel particles during further processing to the fuel element. Furthermore, the heat-shrinking of this Pyrocarbon layer causes a pressure pre-load on the SiC-layer under the influence of fast neutrons and thereby reduces the tensile stresses in it.

– Graphite matrix

The graphite matrix forms the mechanical fuel element construction, in which the coated particles are embedded. In order to avoid local over-heating a uniform particle distribution is targeted in the inner fuel-containing zone.

The matrix takes up the external forces, which act on the BE. It must therefore possess an adequate fracture and impact strength. In order to ensure a perfect transportability and operability, the graphite material must also have a sufficient dimensional stability (*e.g.*, against re-compression, swelling).

The high pressure-pressing of the BE spheres achieves a high density, which is necessary to adjust the adequate strength, thermal conductivity, and carbon quantity (moderation and heat capacity) in the fuel element.

In order to prevent an extremely high quantity of fission products through Uranium fission in the graphite matrix, which would then be released unrestricted from the fuel element, the Uranium contamination in the graphite matrix is to be kept low.

The external fuel-free shell from the same matrix material serves the protection of the coated particles against mechanical and chemical interactions in case of handling and during operation (mechanical damage, wear, corrosion).

Design

The essential function of the fuel elements is to generate thermal energy through fission of the contained nuclear fuel. In order to limit the loads for the fuel elements or the adjacent components that occur due to indispensable handling, operation and possible failures, a series of safety-technological requirements for the design of the fuel elements have to be adhered to so that the crucial safety objectives are achieved. In the following we deal in detail with the loads and the requirement resulting from it and describe, how these requirements are fulfilled.

– Fission product release

In case of the HTR-module-concept the coatings of the fuel particles of the fuel elements present the essential barrier against the release of the fission products. We deal with the release mechanisms as well as the quantification of the retention capability for the specified operation in Section 3.3.1 and for failures in Section 5.8.1 in more detail.

– Mechanical strength

The strength requirements for the fuel elements are established via the drop strength and the crushing strength /U 2.5.2–2/.

The drop strength is defined as the number of falls of a fuel element from 4 m height on a graphite spherical bed that the fuel elements can withstand without significant dimensional deviations (chippings, breakage). Here, a minimum drop strength of 50 falls is required.

The crushing strength is defined as the force, which must be exerted until a BE-sphere breaks between two parallel steel plates. The minimum required crushing strength is indicated by the applicant as 18 kN.

It was shown by the applicant through drop tests and crushing tests that these minimum requirements of the fuel elements planned for the HTR-module are definitively observed /U 2.5.2–2/.

Furthermore the applicant indicates that the strength properties of the BE during the operation (*e.g.*, through radiation or corrosion) do not change considerably.

The applicant has not indicated quantitative values for the loads that act on the fuel elements.

In our opinion, the mechanical loads of the fuel elements occur as

- Dynamic loads in case of handling, *e.g.*, in case of dispatch and in case of the loading and unloading of the core,
- Static loads due to fuel element pressure in the core as well as in individual components in case of handling.

As the operation of predecessor installations (AVR, THTR-300) has shown, the devices for the BE dispatch and handling can be dimensioned in such a way that the loads do not exceed the established minimum strengths of the fuel elements. We deal with this in Section 2.7.3.

- Fuel element-corrosion

Graphitic materials react basically with oxidizing media, *e.g.*, with impurities of the cooling gas such as water, CO_2 or atmospheric oxygen. These reactions have the following effects:

- Decrease of the external diameter through surface wear of material off the fuel-free shell.
- Decrease of the strength through "volume-reaction", *i.e.* inside of the fuel element-pores.
- Formation of ignitable mixtures in the coolant (in case of additional oxygen entry) in case of adequately high concentrations of the reaction products (CO, H_2) corresponding to high reaction yields.

Due to these possible consequences, the corrosion resistance of the fuel elements is specified and controlled in order to keep the effects on the transportability of the fuel elements or on the safety of the installation within tolerable limits with respect to the operating conditions (temperatures, cooling gas impurities) or failures.

The corrosion resistance of the fuel elements is specified with a admissible maximum value for the reaction rate in a standardized oxidation test. The applicant shows with the help of tests on AVR- and THTR-fuel elements through different heat treatments of the graphite during the production that this maximum value can be complied with.

During operation in accordance with the design, the maximum impurities of the coolant with H_2O are limited to ≤ 0.1 vpm and with CO to ≤ 0.5 vpm. According to the applicant's statement the corrosion rates resulting from this are low.

The corrosion caused by failures– the rupture of the hot steam generator tube with water entry into the primary circuit is the determining factor for the design –is dealt with in Section 5.4.3.

In our opinion, it can be ensured to a sufficient degree for the specified operation that no significantly high corrosion of the fuel elements occurs by specifying a maximum corrosion rate for the manufacture of the fuel elements and by monitoring of the oxidizing impurities of the coolant gas with the given maximum values. Therefore we also do not expect a significant reduction of the external diameter or any significant decrease of the strength of the fuel elements. Also in our opinion, the formation of ignitable mixtures can be excluded due to the low corroding masses during the intended operation according to the design.

– Dimensional stability

The transport safety of the fuel elements calls for the compliance with a narrow tolerance range for the fuel element diameter. Changes in the external measurement of the BE during the operation occur due to

– Radiation with fast neutrons, and
– Mechanical wear.

Due to the radiation initially a shrinkage (diameter decrease) of the BE appears, which with continuous radiation changes over to a swelling (diameter increase). Due to the anisotropy of the graphite lattice or the graphite mono-crystal, the extent of this fluence-dependent dimensional change depends on the orientation of the graphite crystals in the BE-matrix.

The mechanical wear of the external fuel-free shell of the BE thus results from the multiple pass through the reactor and the associated long travel distance.

The applicant shows in the /U 2.5.2–2/ document with the help of test results that were obtained for fuel elements exposed in the AVR manufactured from the same graphite-material planned for the HTR-module that only minor dimensional changes, which do not affect the transportability of the fuel elements are produced in case of the fluences to be expected in the HTR-module.

The applicant plans, to establish the maximum admissible anisotropy of the matrix graphite and to specify the wear resistance during the manufacture of the fuel elements.

> Based on the post-radiation examinations of different test element types as well as AVR-fuel elements, which, in case of to-be-expected fluences, show a maximum diameter reduction of approximately 2%, we come to the conclusion that a sufficient dimensional stability of the BE is achievable with the intended graphite materials. In addition, there exists a possibility of removing fuel elements whose diameters fall below the specified value during the cyclic process.

Manufacture

The applicant has described which production processes are planned for the manufacture of fuel elements for the HTR-module-reactor and which quality control measures are established at the manufacturers for guaranteeing a constant quality level. Based on that, the procedure development is largely concluded /U 2.5.2–3/.

At present experience exists regarding the production and the use of around 57,000 fuel elements of the planned type in the AVR. In addition, extensive experience is available for individual steps in the THTR-fuel element production. Furthermore, fuel elements and fuel test samples with a total of about 212,000 coated fuel particles were produced for radiation experiments in different materials testing reactors.

Inasmuch as higher production units are to be adapted for individual procedural steps for introduction of mass production, basically no procedural changes should be made. The guarantee of constant product quality both in case of the necessary adaptations as well as in case of the future production is done through a detailed specification of the product properties and control of these properties within the framework of the established quality control system.

> The positive guarantee of the product properties – especially the fission product retention – achieves a design-specific significance in case of the assessment of the HTR two-module reactor. With the help of the production of the fuel elements and the radiation test samples for the radiation experiments as well as the production of comparable AVR-fuel elements, it is demonstrated, in our opinion, that fuel elements with the product properties required for the KTR module reactor can be produced. Furthermore we are of the opinion that in case of transition to mass production the compliance of the required fuel element properties can be guaranteed with the quality control system and the production and test procedures established in it as well as the remaining organizational and technical quality control measures.

2.5.2.2 Moderator and Absorber Elements

In addition to the modified primary core/start-up phase fuel elements, moderator elements and absorber elements are also employed for the physical neutron compensation of the fission product inventory of the equilibrium core, which has not yet built up in the initial core or in the start-up phase, respectively.

The moderator elements are spheres of 60 mm diameter made from nuclear-purity electro-graphite. The requirements for mechanical strength and corrosion resistance are the same as those for the fuel elements.

The absorber elements are produced from the same materials and by the same production steps as the fuel elements. The inner zone of 50 mm diameter however contains neutron-absorbing substances, such as Hafnium carbide and/ or Boron carbide coated with pyrolytically separated Carbon and Silicium carbide, instead of the coated fuel particles. The requirements for the mechanical strength and corrosion resistance of the absorber elements are the same as those for the fuel elements. The other requirements with respect to temperature stability correspond appropriately to those of the fuel elements.

> Since the moderator elements and the absorber elements are produced by similar process steps and largely use the same materials as the fuel elements, we are of the opinion that the requirements placed on the fuel elements in respect of dimensional stability, breaking strength, corrosion and temperature stability can also be maintained by the moderator and absorber elements. Besides, experience from manufacturing and use of moderator and absorber elements is also available from AVR and THTR.

2.5.3 Core Internal Fittings

2.5.3.1 Ceramic Internal Fittings

Constructive Design

A detailed description of the ceramic internal fittings is given in the Function and Design Description /2.5.3–1/. More layout details can be referred to in Supplemental Documents on Heat Insulation /2.5.3–2. An important principle of design that has been tested in other HTR systems as, *e.g.*, in the case of the AVR and THTR-300, is the partitioning of the structure into individual blocks that are joined to each other in axial and radial directions either by means of dowels, tongue and groove, or through dovetail connections. By partitioning into individual structures, obstructions to elongations caused by operational temperature differences or radiation effects are avoided and thus generation of major macroscopic stresses are prevented.

Bypasses for the cooling gas flow result from the gaps between the individual internal fittings, directed from outside towards the inside of the core. Performed calculations, confirmed by results from experiments, show that the deficit of directed cooling gas restricted by the gap can be limited to a few percent of the entire cooling gas flow by appropriate dimensioning of the constructive elements.

The following design features, which already form the basis of the THTR-300 system, are also provided for the HTR module.

- Structuring of the side reflector in individual columns, which can carry out thermal shifts in axial direction, independently of each other.

- Use of graphitic materials with suitable physical neutron properties with simultaneous major heat conductivity for the inner areas of roof, sides and floor reflectors, oriented towards the core. Carbon rocks are provided for the external part of the reflector, which have a much lower thermal conductivity than graphite and thus protect the adjacent metallic internal fittings from inadmissible temperature load. For reducing the activation of the reactor pressure tank, the metallic core internal fittings and the hot gas pipeline, the floor position and the neighboring area of the side reflector are treated with boron.

- An independent support for the floor reflectors, separated from the side reflector, so that both components do not hamper each other in case of thermal shifts.

- Restricting the expected thermal shifts in side reflector blocks outwards in the radial direction and in azimuthal direction through additionally placed metallic structural elements.

- Use of "failure depressions" on the core-side internal surfaces for improving the flow behavior of the fuel element spheres.

Meanwhile we assess the usage of design features as proven for other HTR systems as positive.

In view of the expected operational or interference load-related thermal shifts of the reflector blocks, we are convinced through our own relevant calculations that these shifts do not cause any inadmissible changes in shape and dimensions of the core enclosure and channels for the control and switch-off elements and cooling gas flows and that the integrity of the ceramic installations and the bordering metallic inclusions are not adversely hampered by them.

The temperature shifts determined by us are based on the radial and axial temperature paths /U 1, U 2.5.3–3/ as determined by the applicant with the help of the THERMIX computing program for representative operating conditions and failures, which we have checked and can confirm. The self-regulating temperatures are monitored with the help of fixed thermo-elements at several vertical positions on the internal area of the side reflectors in the floor and ceiling reflectors as well as in the cooling gas area/U 2.5.3–4/.

We consider the intended boron treatment for the ceramic core inclusions in the floor and in the external side reflector area as a suitable means for the activation reduction of the reactor pressure tank, metallic core inclusions, and the heating gas line.

Strength design

The applicant has given a detailed description of the strength design of the ceramic internal fittings, above all that of the maximum stressed side reflectors, in /U 2.5.3–5/. The said document provides an overview of:

– The strength and physical properties of two representative types of graphite for the un-radiated condition and for one of the two types for the radiated condition.

– The to-be-expected pressure, temperature and radiation loads under operative and failure conditions.

- Stress distribution in the maximum stressed areas of the side reflector under operative and failure conditions, determined with the help of a computing program specially developed for reactor internal fittings made from graphitic materials.
- The strength design as provided by the applicant for evaluation of the determined stresses.

The expected pressure loads for the side and floor reflector due to sphere fill of fuel elements can be deduced from the Documentation for Strength-based Assessment /U 2.5.3–5/. The silo theory /L 34/ was used for calculation. We can confirm the presented results by a calculation carried out independently of us. For additional forces resulting from the pebble bed compression in non-stationary operation due to thermally dependent expansion differentials, the results from experimental model investigations /U 2.5.3–5/ are available. We have no objections regarding the extent of the mentioned additional forces, which are superimposed on those from the silo theory. Overall, the resultant stresses for side and floor reflector are small in comparison with the admissible stress. The corresponding order of magnitude is 1 N/mm^2 and below.

Higher stresses result from operative temperature gradients or that from failures according to calculations performed by the applicant. We concur with the determination and the results of the temperature curves in radial and axial direction on basis of the temperature gradients. To calculate the resulting stresses, the applicant used the Finite Element Calculation Program JAGUAR /L 93/. We also have no objections to the use of this generally recognized computing program that is especially developed for use with RDB graphitic internal fittings and for the mechanical stresses determined thereby.

The results can be summarized and evaluated as follows:

Expected operative temperature gradients result in stresses of a few N/mm^2; in the most adverse case (incursion of small spherical shutdown elements) stresses of just under 10 N/mm^2 result. Slightly higher stresses for the reflector parts result due to radiation effects. It can be deduced from the stress curves determined by the applicant and plotted dependent on dosage, temperature and direction, that at points particularly important for safety, for example at the bridge of the long hole for the spherical shutdown elements, one can reckon with a resultant total stress of maximum about 1 N/mm^2 under the /U 2.5.3–5/ irradiated state. The maximum stress results from the applicant's calculations for the mentioned position, under failure conditions (incursion of small spherical shutdown elements) at 11.3 N/mm^2. It appears worth mentioning in this context that comparably high maximum stresses were determined through calculations carried out with another computing program for the side reflector of the THTR 30 under similar stresses due to temperature differences and neutron irradiation. We have expressed our opinion on the neutron flows used for the calculation of the stresses for the graphitic inclusions of the HTR internal fittings in Section 3.1 and confirmed the values used.

For assessing the determined stresses, the applicant has developed an assessment concept. Using this assessment concept, which is further explained below, a safety factor of approximately 2 is obtained for the above-mentioned to-be-expected total stresses and a nominal value of 22.1 N/mm^2 is derived for the bending strength of a currently available reactor graphite. It must also be taken into consideration here that the strength data for graphite compared to the here considered un-irradiated initial state as the basis, increases through neutron irradiation as well as with increasing temperature within the temperature and dose ranges to be considered, and thereby also raise the mentioned safety factor.

The applicant has not submitted any separate calculations of the mechanical stresses and strains to be expected for the side reflector in EVA failures. In our view, core and ceramic installations need not be regarded as expressly vibratory structures and can therefore also be regarded as monolithic in respect to EVA loads. A conservatively calculated estimate performed by us of relative movements due to safety earthquakes and plane crashes resulted in only minor stresses and strains for all considered structural elements; they are also covered by the above mentioned maximum stresses.

The above-mentioned design by the applicant regarding the strength assessment of the determined stresses is demonstrated by the use of following terms:

- Use of the criterion of peak elongation energy for linking the spatial stress condition with the strength characteristic values determined in single axis stress or pressure tests.
- Definition of a calculation factor (nominal strengths) for bending, stress and compression stresses of the material used by the following equation:

  Calculation factor = Mean value of the readings minus double the standard deviation

- Safety factor S for defining the allowable stresses:
  - S = 1.5 to 3.0 for protection against the total stress, and
  - S = 4.0 for protection against primary stresses.

We consider the applicant's suggestion given above for assessing the total stress determined for the side reflector of the HTR and of the primary stress to be safety-compatible. The indicated method is in agreement with the knowledge obtained from the BMFT research project "HTR design criteria" for the graphitic reactor inclusions /L 94/. We agree that the safety factor S is specified in the respective actual application within the mentioned range. The precise determination will depend, according to the form of the stress distribution, upon the possible tolerability of locally restricted crack formation and probability of occurrence of the respectively considered load type.

The applicant points out in the technical documentation /U 2.5.3–5/ that in case of the graphitic inclusions it suffices to regard the considered stresses as quasi-static, because stress change tests on representative graphite samples have shown that their strength characteristic values do not decline to a considerable extent in the range of the expected load change /L 95/. Because of the said test results, we have no objections to the quasi-static treatment of the strength design of the graphitic inclusions.

The above-described design for determination and evaluation of the mechanical stress on structural elements of the graphitic side reflector is also transferable to that of the graphitic floor and ceiling reflectors. The total stresses to be expected are lower than for the side reflector. For the carbon rocks arranged on the outside of the graphitic floor, side and ceiling reflectors, the applicant has not put forth any specific strength design in view of the stresses and strains to be expected, which are considerably lower in comparison with the graphitic internal fittings. We agree to that. The considerably lower stresses of the carbon rocks results from their being present in areas in which the neutron flow density against that of the core-side part of the graphitic reflector is lower by about three orders of magnitude. The temperature gradients to be expected in the operative and failures are also considerably smaller.

A synopsis of strength-related material properties is given in documents /U 2.5.3–2/ and /U 2.5.3–6/ with examples of several types of graphite bricks and carbon rocks. This shows that the compression strength of the carbon rocks is at least equal to that of graphite bricks. As compression stresses are mainly to be reckoned with in case of carbon rocks, we expect no problems for transferring radially or axially directed forces from graphite bricks to carbon rocks.

A reduction in strength of ceramic installations due to corrosion that has to be considered for the design can be excluded based on well-found reasons. From experimental investigations /L 156/ we understand that the weight loss in relation to the original weight is less than 1% under very conservative conditions (830°C, water vapor content 1 vpm) after 30 years life expectancy. The highest fuel element temperatures occur at the lower end of the discharge. The calculated surface temperature of the fuel elements is 812°C. The water vapor concentration in circulation is kept under 0.1 vpm throughout the Helium cleaning system. We see from the Document on Reflector Graphite /U 2.5.3-6/ that with a conservative estimate of a weight loss of 0.5%, a strength reduction of less than 5% can be expected. A similar amount of strength reduction (0.3% weight loss) can be expected as per the Safety Report as a consequence of the postulated failure "Tearing-off of a Steam-Producing Heating Pipe."

In view of the Wigner Effect, which in principle yields the possibility of a spontaneous release of the Wigner energy stored in the crystal lattice as a result of an uncontrolled self-heating, it is to be checked whether this has any safety-technology related consequences for the HTR module.

The same question arose in the context of approval for commissioning of the THTR-300 in connection with locating the causes of an accident, which occurred in one of the two graphite-moderated, air-cooled reactor blocks in Wind scale on October 10, 1957.

Detailed investigations /L 76/ have shown that there no safety regulations are necessary for the THR-300 as a precautionary restriction of output or additional measures against an unchecked release of Wigner energy. The temperature range between 150°C and 200°C – critical for saving energy – is bypassed or traversed only briefly due to the operationally provided temperature control for stationary and non-stationary operation.

It can also be seen from the operational data of the HTR module that the temperature range critical for energy saving is bypassed during stationary operation. During start-up or shutdown, the critical temperature range is traversed for such a transitory period only that no noteworthy energy storage is possible. The storage of appreciable Wigner energy would need an operation within a critical temperature range of below 200°C for several months at the flow density available for the HTR module. For the above-mentioned reasons, we do not regard additional specifications necessary as precautions against a safety-related storage of Wigner-energy.

Quality Control of the Materials

Besides the classification concept/U 2.3–1/, the applicant has provided an appropriate quality control for the functional requirements for the ceramic installations and thus the Criterion 2.1 of the Safety Criteria /L 7/ is met.

For safety calculations of the side reflector, the applicant has, for example, presented the graphite strength data of AS-1RS category (manufacturer: SIGRI, Meitingen) as the basis for the un-radiated and radiated state /U 2.5.3–6/. The applicant contemplates using more cost-effective graphite classes, for example the graphite of ASR-1RG category (manufacturer: SIGRI, Meitingen) for ranges with lower strength stress, for example for the ceiling reflectors and side reflectors, whose strength is somewhat lower than that of the graphite of ASR-1RS class.

> We have no objection to the method described by the applicant in the reports /U 2.5.3–5/ and /U 2.5.3–6/, by which the specification values of the strength data are brought in line with that of the respectively to-be-expected maximal stress. In view of the published results on hand of statistical evaluations of the material data measured for other HTR for the acceptance of graphite and charcoal and the distributions presented by the applicant in the /U 2.5.3–5/ report for failure probability, we have no fears that the requisite material properties and the analogous requisite dimensional tolerances of graphite and charcoal can be maintained sufficiently.

Recurring Tests

According to the Safety Report, visual inspections for checking integrity and surface quality in the switched-off condition of the system are in principle possible. For example, it is possible to visually inspect the long holes of the spherical shutdown elements by means of a video camera from the ceiling area all the way to the floor structure.

Besides, in emergency cases it is also possible, as per the applicant's information, to inspect the internal surfaces of side reflectors, core floor and fuel element drainpipe, after complete evacuation of the core, with the help of a video camera and to carry out the requisite repair jobs if required.

> On the basis of our tests, we can confirm that required recurring inspections can be carried out with the switched-off system.

Summarizing Assessment

> The present concept of the ceramic installations meets the operational requirements, which arise from the problem definition and the failure conditions to be expected. This is particularly applicable for the requirements named for the inclusions in the BMI safety criteria (3.3).

2.5.3.2 Metallic Installations

Constructive Design

The metallic installations consist of the following components, according to the safety regulations and the design description /U 2.5.3–1/:

- Core vessel with guides and supports,
- Floor structure with floor plate,
- Thermal cover shield,
- Overturning locks for the ceiling reflector with ceiling reflector plate, and
- Fuel element drainpipe (metallic part).

- Core Vessel with Guides and Supports

  The core vessel, together with other components, serves as carrier and support for the ceramic installations and ensures gas exclusion (air exclusion) in case of opened reactor pressure vessel for maintenance activities. It is built as a cylindrical shell, consisting of individual sections, which are welded to one another. Some sections are reinforced. They are placed at various levels and serve to accept guide elements for azimuthal guidance of the side reflector. Steel sections are placed for supporting the ceramic installations distributed evenly in the floor and ceiling area of the core shell, outside the core zone at the periphery. At the top end of the core vessel there is a flange joint with metal sealing elements for accepting the thermal cover shield.

At the lower end of the core vessel, there is another flange, which is designed as a carrier flange and serves as a seat for the floor structure.

In the lower area of the core vessel, laterally, there is a flange joint for the connection of the heating gas line.

One of the reinforced sections is equipped with an external flange, which serves for bearing (supporting) the core vessel in the reactor pressure tank.

– Floor Structure with Floor Plate

The floor structure is designed as a welded construction. It is designed as a stiffened truncated cone with top and bottom cover plate. The stiffening consists of radially arranged steps. In the lower cover plate, there are openings for mounting the discharge vessel of the small spherical shutdown system integrated in the floor structure, as well as the pertinent pipelines. The floor plate is positioned with radially arranged steps on the top cover plate. It has blind holes, which serve as dowelling joint for the ceramic core structure. By means of a circumferential ring, both plates are connected and are made watertight. The floor plate has holes for cold gas conveyance and discharge of small spherical shutdown elements. It also has a central hole, to which a pipe section fits, which forms the metallic part of the fuel element discharge pipe. The top and lower cover plate similarly have a central drill hole, which takes in the fuel element supply and serves to guide the cooling gas.

– Thermal Cover Shield

The thermal cover shield is designed as a cast iron plate. This plate has holes for the small spherical shutdown units, reflector bar system and the fuel element feeding tube. On the plate, guide elements are placed in the peripheral direction, flush with the reactor pressure vessel internal walls, which are supposed to secure the core vessel against seismic stresses.

– Tipping Locks for the Ceiling Reflector with Ceiling Reflector Plate

Segmented casting blocks are used as tipping lock for the ceiling reflector segments. The ingots have holes for the passage of small spherical shutdown elements and the reflector bars as well as inspection openings. The ingots are dowelled with the side reflector.

The ceiling reflector plate is designed as a round plate, which is slit radially corresponding to the ingots pitch and joined to it with bolted joints. The adaptation of the plate is to enable vertical relative movements of the individual columns of the reflector. It also has holes in the ingots for the passage of small spherical shutdown elements and the reflector bars as well as inspection openings.

– Fuel Element Drainpipe (Metallic Part).

The metallic part of the fuel element discharge pipe consists of a pipe section welded to the floor plate and the connecting thick walled cast rings. The cast rings are plugged. The rings are supported on the reactor pressure tank.

In the design of the individual components of the metallic core installations, proven construction principles have been observed for vessel and steel construction. The connection of semi-products for components is done mainly by welding. Bolted joints are used only for the parts used for carrying and transmitting horizontal forces. The core vessel and the floor structure are designed to be suitable functionally and for force-flow.

Strength Design

The stresses from stationary and non-stationary operative processes for actual operation and the stresses from failures due to non-stationary temperature processes as well as external effects must be considered as the basis for the strength design of the metallic inclusions, as per design description /U 2.5.3–1/.

The principal mentioned stresses on the metallic installations are

- Weight and silo forces from the ceramic installations,
- Seismic stresses, temperature stresses (stationary and non-stationary), and
- Pressure differences in the primary coolants.

The design temperature for the core vessel is 500°C. For the individual components, the materials selected are as follows:

– Core vessel	15 Mo 3 or 10 CrMo 9 10
– Floor structure with floor plate	15 Mo 3 or 10 CrMo 9 10
– Thermal cover shield	GGG-40
– Side reflector supports	
– Guides	GGG-40
– Support rings	15 Mo 3 or 10 CrMo 9 10
– Ceiling reflector plate	15 Mo 3 or 15 MnNi 63
– Tilt locks of ceiling reflector segment	GGG-40
– Fuel element drainpipe	GGG-40

The temperature stress is the most important design factor for the strength-relevant design of the metallic core installations. To protect the reactor pressure vessel and the metallic core installations from unreasonably high temperature stresses during the actual operation and in case of failures, a series of design measures is contemplated. Using these measures, the failure temperature of the metallic core installations can be limited to a maximum of 490°C /U 2.5.3–2/. Our temperature calculations also confirm the selected design temperature of 500°C. At this temperature, the selected materials still possesses sufficient creep strength value. From this perspective, we consider the proof of satisfactory strength design of the metallic core installations for the actual operation and failure cases to be achievable.

Quality Control of the Materials

For the metallic core installations, the classification concept /U 2.3–1/ provides for a specification of the quality requirements for materials as well their processing, corresponding to the functional requirements for the installations. There are no reservations regarding this method.

Recurring tests

As per the Safety Report, the space above the core vessel is accessible for visual inspection for an open reactor pressure vessel. It is also possible to visually check the external shell of the core vessel up to the bearing on the reactor pressure vessel using a video camera. According to the applicant, it is also possible, in an emergency, to inspect the metallic part of the fuel element drainpipe using a video camera, after completely evacuating the core.

> In our opinion, recurring inspections can be carried out to the required extent with the reactor switched off.

Assessment Summary

> In our opinion, the presented concept of the metallic installations meets the requirements that arise from the problem definition and the to-be-expected stresses during operational and failure-related loads. This is particularly applicable to maintaining the core geometry in view of the sufficient post heat transfer and safe shutdown of the reactor.

2.5.4 Control and Shutdown Devices

2.5.4.1 Shutdown Concept

In this section, we shall consider the presented concept of switching off the HTR module and the task assignment to the individual facilities intended for shutdown. According to the technical documentation for the shutdown concept /U 2.5.4–1/, we can differentiate between

- The shutdown through neutron absorber, and
- The shutdown through interruption in the primary coolant passage.

This concept of the applicant is the basis for the subsequent description of the shutdown concept.

Shutdown through Neutron Absorber

In case of a shutdown through the neutron absorber, fission of the fuel material is interrupted by interposing neutron-absorbing materials through holes in the side reflectors. Two shutdown devices are provided for this kind of shutdown, which are designated as “first” and “second shutdown devices”.

The first shutdown device consists of the reflector rod system. According to the Safety Report, this system is designed in such a way that it can sufficiently quickly render the reactor powerless and sub-critical during actual operation as well as keep the reactor in the “hot sub-critical” state for a sufficiently long time. The failure of the most reactivity-effective reflector rod was also taken into consideration while designing.

The reactor protection system results from control of the first shutdown device together with the shutdown of the primary system pump. The reflector system is also called "hot shutdown system" due to its property of shutting down of the reactor in the "hot uncritical" state.

The second shutdown device consists of the small pebble shutdown system. This system is designed in such a way that it is in a position to make the reactor uncritical from all control operational conditions that do not require quick reactivity changes and to keep it sub-critical for as long as required at the lowest temperature of 50°C /U 2.5.4–1/. The small pebble shutdown system is therefore also called cold shutdown system. No automatic control of this shutdown device has been provided for. If required, it is triggered manually.

Shutdown by Interrupting the Coolant Flow Rate

One option to reactor shutdown is to simply interrupt the primary coolant flow rate by shutting down the primary circulation pump. As per this observation, the breakage of the reflector rods is not considered; first there is no interruption in the power generation. On account of the missing operative heat supply through the secondary circuit the core temperature rises and then due to the negative temperature coefficient of the reactivity, sub-criticality of the reactor occurs at a higher temperature level. However, this shutdown effect is not used in the framework of the proposed shutdown concept, as the pump shutdown and the reflector rod breakage are always automatically excited together.

We have considered the following rules and guidelines as evaluation criteria:

- Safety criteria for nuclear power plants /L 6/
  Criterion 5.3: Devices for Control and Shutdown of Nuclear Reactor
- KTA-Rule 3103: Shutdown of Light-water Reactors /L 45/
  Chapter 3: Tasks of the Shutdown Systems

The following main requirements result from the evaluation criteria:

There must be two shutdown devices available, which in their entirety may

- Transfer the reactor into the sub-critical, powerless condition, and
- Keep the reactor continuously in a sub-critical condition under the most unfavorable conditions.

To meet these safety regulation functions, both the shutdown devices must be constructed independently and different from each other. A shutdown device designated as a quick shutdown system must by itself be in the position to make the reactor powerless and sub-critical under control operation as well as during failures in a quick manner and be capable of retaining it in this state for a sufficiently long time. Another shutdown device must be able to make the reactor powerless and sub-critical from all conditions of the control operation that do not demand any change in reactivity.

The shutdown device designated as quick shutdown system is to be assigned to the safety system. If the two above-mentioned safety regulations cannot be fulfilled by this shutdown device alone on the occurrence of failures and in anomalous operating conditions, then further requisite shutdown device is also to be assigned to the safety system. The time of their employment as well as the decision, as to whether their use can be manually executed or must be triggered automatically, are to be defined on basis of the sequence of events analysis.

The shutdown device designated as quick shutdown system must be able to meet its safety regulation even in case of the failure of the most reactivity-effective control element in such a manner that the respective specified limit value of the reactor system cannot be exceeded.

If the shutdown devices are wholly or partly used for operational control, it must be ensured that a sufficient effectiveness reserve is available for the safety regulations at any time and that the safety regulations have priority over the operational requirements. Further, a functional operational control or a control system error must not compromise the safety regulation of the shutdown device.

The shutdown concept shall be evaluated below in view of the availability of the devices required for shutdown of a reactor in conformance with the evaluation criteria or their assignment in respect of the tasks to be fulfilled. The evaluation of the efficiency of the shutdown devices is performed in Section 2.5.5 "Shutdown Security" that of the control in Section 2.13.2 "Reactor Protection System".

With the proposed concept, providing for the reflector rod system as the first shutdown device and the small pebble shutdown system as the second shutdown system, the demand for two shutdown systems is met.

The demand that both the shutdown systems must be structured independently and different from each other is met through the following design features:

- The reflector rod system and the small pebble system are constructed differently,
- Both systems have neither common components nor common help systems,
- The absorber materials are used in different forms in the two systems.

The demand for a quick shutdown system as a quick working shutdown device, which is by itself in a position to quickly make the reactor uncritical from all conceivable failures, is met by providing the reflector rod system in conjunction with the shutdown of common pump. Further details can be obtained from the Sections 2.5.4.2 and 2.5.5.5.

In our evaluation of the shutdown concept, we have not taken any credit away from the shutdown option by interruption of the primary coolant flow rate as proposed by the applicant. We are however of the opinion that the possibility of interrupting the power release through fission only with a blower shutdown represents an additional safety benefit, which is available in case of high temperature reactors with negative back-coupling properties and which was recorded in the AVR for the so-called rod clamping trial.

In summary, we establish that the proposed shutdown concept is suitable, with respect to the available devices as well as their assignment to the tasks to be fulfilled, to meet the requirements resulting from the decision criteria.

2.5.4.2 Reflector Rod System

Objective

The reflector rod system forms the first shutdown device in the shutdown concept of the HTR module and is designed as a quick shutdown system with the blower shutdown for transferring the reactor into the "hot sub-critical" state even on failure of the effective rod during control operation and in case of conceivable failures and keeping it like that for a sufficiently long period.

Besides this safety regulation, the reflector rods meet the operative tasks of reactivity control in power operation of the HTR module.

Construction, Design and Layout

The reflector rod system consists of six reflector rods with the corresponding drives and control technology. The reflector rods are moved up and down by the drives hanging freely in the side reflector holes. The six reflector rod drives are arranged above the so-called thermal cover shield within the reactor pressure tank, the link to the rod holes being realized through a metallic guide pole. Guides are not provided for in rod holes /U 2.5.3–1, U 2.5.4–3/.

The important components of the reflector rod drive, according to the Design and Functional Description /U 2.5.4–3/, are:

- Electrical motor,
- Planetary gear with line-side bevel gear or cylindrical gear set between chain socket and electrical motor,
- Round link chain as linking element between drive and reflector rod,
- Eddy current brake with permanent magnets on second motor shaft end,
- Damping element on drive-side end of the round link chain,
- Position indication with measuring gear and synchro-transformer as well as magnetic switch for upper and lower end position of the reflector rod.

Thereby, the round link chain subassembly, damping element and line side gear box together with the holder device for the rod mounting, the sealing system for maintenance as well as inner shielding are housed in an approximately 3000 mm long tube-shaped container.

The ambient temperature indicated for the drives is about 150°C. Design pressures are not indicated because the components are arranged within the reactor pressure tank. The applicant has given the pressure difference between blower side and pressure side as the differential pressure.

In specification /U 2.5.4–3/ the following are listed as important characteristics for the drive

- The control velocity at approximately 10 mm/s,
- The shutdown velocity at approximately 0.5 m/s,

Each reflector rod consists of ten individual links of approximately 500 mm in length. The absorber of sintered B_4C rings is located between two co-axial sleeve tubes also made from the material used for THTR (Mat. No. 1.4981). The gaps between the sleeve tubes and the B_4C rings are measured in such a way that no forces are generated through swelling of the B_4C or through differential thermal elongation. The main data are indicated in Table 2.5.4–1. The individual links of the reflector rods are connected with each hanging above one another by means of a central rod.

The top end is connected with the drive rod, which creates the mechanically locked lock to the drive.

The lower rod end has a taper to center itself in case of a rod breakage due to a mechanical failure of the drive parts in the shock absorber. This shock absorber is located in the lower end of the reflector rod bore in the ceramic core structure higher than the core floor. For protecting the ceramic core structure, the shock absorber consisting of a compression tube is enveloped by a maintenance tube.

Table 2.5.4–1: Design data of the reflector rods

Number per module	6
Dimensions (mm) – B_4C-Rings External diameter of the reflector rod – Bore diameter in the ceramic side reflector – Total length – Total active absorber length	 ø 100/76 × 160 ø 105 mm ø 130 mm 5,280 mm 9 links of 525 mm 1 link of 555 mm 4,800 mm
Materials – Absorber – Sleeve tube, suspension	 B_4C-Rings X8CrNiMoNb 16 16 (Mat-No. 1.4981) (THTR sleeve tube material)
Total weight	approximately 104 kg
Design temperature of the sleeve tube	650°C

Functioning During Control Operation and During Failures

The reflector rod is moved up and down in the side reflector bores with a velocity of approximately 10 m per second by the reflector rod drive, which consists of an electrical motor, gearbox, chain socket. During the upward movements of the rod, the chain is deposited with a loosening tool into the open chain box. During downward movement, the chain is again taken from it. The two-phase motor works as an actuator, which works with a constant voltage in one coil and a variable voltage on the second coil. The modes provided are "Control Mode" and "Emergency Control Mode". In case of a quick shutdown of the reactor, the supply voltages are switched off on all poles so that the rods fall into the reflector bores under gravity. The maximal falling path of the fully traversed position up to the lower end position, approximately 1000 mm below core center, is approximately 6800 mm. The falling-in velocity is limited by the eddy current brake to approximately 0.5 meters per second. The damping element on the drive input side chain end absorbs the kinetic energy of the falling rod and the rotating masses in the lower end position.

A shock absorber is provided in the reflector rod bore for the failure of the link between reflector rod and chain or the tearing of the drive-side chain end of the damping device. The measurement of the upsetting tube absorbing the falling energy is planned for the assumed breakage of the whole rod.

The six reflector rods can be moved individually by the control, though this is not provided for in the undisturbed operation. The motors are supplied with two-phase 230 Volts AC voltage. One thyristor per drive controls the movements towards the target value indications of the power control. The descent depth of the rods is limited to ensure the requisite shutdown reactivity.

The control of the quick shutdown of the reactor takes place in three channels from the reactor protection system by means of a 2 × 6 contact system. A 2v3 bias current principle is used. In a Resa case, the power supply of the motors is thereby interrupted.

The freely suspended reflector rods in the side reflector bores are cooled inside and outside by a bypass current of the main cooling gas current. The coolant gas enters at a temperature of 250°C into the bore in the cold gas collection room area in the ceiling reflector and flows through the reflector rods from top and bottom. The cooling gas exit is in the core floor area /2.5.4–3/.

The arrangement and layout of the reflector rods and of the drives specified is so that maintenance and repair jobs as well as configuration of rods and drives is possible.

The following devices are named for this purpose:

- Manual drive of the electrical motors,
- Locking of the reflector rod in configuration setting,
- Sealing system between rod suspension and drive housing,
- Coupling between reflector rod and round link chain
- Swivel fixture for rod coupling,
- Slider for preventing air entry in the core area /2.5.4–3/.

The jobs are carried out above the accessible thermal shield. Shields are available.

Inspections and Component Tests

The reflector rod system of the HTR module largely corresponds to that of the THTR-300.

Prototype tests were carried out with the rod drive under THTR specific conditions /2.5.4–4/. It is planned to revert to these tests, for which however extensions of the HTR-module-specific condition are required. These are in particular other temperatures, pressure ratios, radiation exposure, layout, travel and reflector rod weight /2.5.4–5/.

Similarly planned are prototype tests for a 'model precursor' for smooth functioning under closed-reactor conditions having an impact on the environment conditions, on failure and erroneous functioning of individual components, on service life and installation process /2.5.4–5/.

It is planned to carry out accompanying checks on the components of the reflector rod systems of the HTR module. These tests include

- Preliminary tests,
- Material and structural tests,
- Functional tests

and their documentation. Details in this regard should be defined according to the Documents on the Quality Control Measures /2.5.4–5/ within the framework of the constructional planning of the technical inspection and acceptance conditions (specifications).

The concept for periodical tests is also presented in the document /2.5.4–5/. Tests are also planned for the power operation, during standstill as well as for disassembled reflector rod drive.

As an important quality criterion, we have based the BMI Safety Criteria on the safety criterion 5.3. In our opinion, the KTA rule 3103 "Shutdown System of Light-water Reactors" cannot be applied unrestrictedly. Subaspects, such as the always cited design specifications based on safety regulation, can however be the basis for an assessment. This also applies for the RSK guidelines for pressurized water reactors, Section 3.1.2 for IL 101 "Shutdown System." The requirements of the KTA rule 3501 "Design of the Reactor Protection Systems" are valid for the control system requirements.

The safety regulation as per Resa resolution for the reflector rod system stipulates that all the rod positions function even after falling of the reflector rods. This is ensured if the rod falls in on demand to the shutdown position under its own weight, without it being unduly decelerated by the chain or the components. The limiting of the falling velocity is achieved through the intermediate link motor and transmission of the eddy current brake.

Ensuring trouble-free reflector rod falling even over a long period of use with high reliability demands that the components of the reflector rods and the transmissions be designed under the expected operating conditions, such as stress frequency due to acceleration and delay as well as material characteristics and wear and their operative characteristics that are ensured through experiments and evaluation of operational experience.

The analytical assessment of the prototype testing with the THTR components and the planned module-specific trials on a precursor module prototype unit are suitable for providing a sound foundation. The tests and trials presented in the design ensure that the requirements of the 3103 KTA rule for the particularization and definitions provided for within the framework of constructional planning can be met.

The requirements of the rule 3103 KTA can also be met by the accompanying checks, which are particularized and defined within the framework of the constructional planning on the basis of the proposed design and periodical tests with the operational records on the operation, annual fall time inspection, inspection of the position indicator as well as inspections of drive in multi-year cycle.

It is to be understood from the THTR Evaluation Report for Thermal Design /L 77/ that the design for cladding tubes has been successful at 650°C. A complete statement is not included for the reactor operation period, but the use is ensured up to the planned inspection dates.

We are of the opinion that the nominal operating temperature for the cladding tube is adequately well covered for the design temperature of 650 °C, also mentioned in the HTR module.

The applicant has mentioned the following maximum attainable temperatures:

- Failure of the main heat sink 700°C,
- Pressure release failure 860°C

During appraisal, we must differentiate between ensuring the function (Resa) and maintaining the integrity of the parts of the fallen reflector rods. As there are no changed conditions for the Resa from the above-mentioned releasing occurrences, in comparison to the operation as per the rules, so a safe falling of the reflector rod is also ensured in these cases.

After the falling of the rod, the lower two thirds of the reflector rod heats up to over 650°C over a long period. It is therefore necessary to statically discharge forces for the affected parts of the reflector rod under its own weight at higher temperatures. These temperature ranges of up to 700°C or 860°C have not been specified for the THTR cladding tube material of the core rods as per current materials technology. However, in this case we can still reckon with the integrity of the components up to 860°C for the design temperature of 650°C on the basis of the above-mentioned occurrences. Within the framework of constructional planning, proof is to be furnished in such a manner that stressed reflector rods fully comply with the requirements for subsequent operation as per the rules. As the reflector rods can be changed if necessary, this is not relevant to the design.

Boron carbide B_4C in the form of sintered rings is used in the reflector rods for neutron absorption. The following are to be taken into consideration for assessment:

- Swelling under radiation
- Temperature rise due to n-reactions
- Reciprocal mechanical and thermal interactions with the cladding tube material as well as
- burn-ups.

Based on the occurrence as per the THTR evaluation report for the core rods /L 77/ with B_4C and the operational experience with other gas-cooled reactors, no critical problems are visualized for the concept. This is particularly the case because the stresses due to the arrangement of the HTR module reflector rods are overall less than in case of THTR core rods.

An ambient temperature of 150°C in stagnant Helium is indicated for the reflector rod drives for the operation as per the rules. As this mathematically determined steady-state temperature is fraught with uncertainties, this had been adequately considered by selecting a design temperature of 300°C, and this has been given as a supplement to the functional description /U 2.5.4–3/.

The selection of the metallic and non-metallic materials for the reflector rod drives must be done according to the allowable temperatures, from the point of view of strength and wear. The planned rise of the design temperature of the metallic materials for the HTR module drive in comparison to the THTR reflector drive is not to be regarded as problematic. Non-metallic materials, which were used for the THTR drive /L 77/ are not to be used in the same condition for the HTR module drive. So a conversion from oil to solids has been provided for lubricants /2.5.4–5/. Different plastic or a metallic material is being used for the shock absorber. Besides, other insulation materials are planned for electrical components. Based on the testing of components and materials in other areas (*e.g.* Molycote lubricant at an operating temperature of 280°C and glass fiber insulations in the core rod drive of the THTR) we expect no underlying problems.

The fundamental rule of controlling the quick shutdown device has been confirmed in comparable power plants. We do not have any objection to its use.

Due to the high design temperatures and temperatures occurring during operation of electrical engineering components of the HTR module reflector rod drives, there are only a few options for selection for the components and materials to be used. In our opinion, however, a suitable selection and the required qualification are possible.

Servicing, inspection and change of components of the reflector rod system is completely possible with the devices planned for extension work on the reflector rods and the drives. Details in this regard must be defined along with the constructional planning.

We consider the planned reflector rod system of the HTR module as technically feasible based on the submitted documents as well as in comparison with the core rod system reflector rod system, evaluated for the THTR. After appropriate specification of the detail requirements for the manufacture and testing, the basic rules and guidelines can be met.

2.5.4.3 Small Pebble Shutdown System

Objective

According to the shutdown design, the small pebble system functioning as the second shutdown system has to make the reactor sub-critical from all states of operation as per the rules, which do not demand any quick reactivity change and to keep it sub-critical for any length of time at the lowest operating temperature of 50°C. As a supplementary to the quick shutdown with the reflector rod system, the small pebble shutdown system serves to ensure long term maintenance of the "cold sub-critical" state.

The small pebble shutdown system also has to fulfill operational functions of reactivity control in certain operational states, besides this safety regulation function.

Construction, Design and Layout

The small pebble shutdown system is constructed from 18 independent units /2.5.4–6/. A unit principally consists of the following components:

- Storage vessel with lock and cyclone separator,
- Discharge vessel,
- Transport line and transport gas feedback,
- Small pebble shutdown elements.

The transport gas fittings and the transport gas bypass fittings of six units each are combined with one fittings block each. For each unit, there is a speed-controlled material transport blower, which is connected in-line to the dust separator. The main design data are listed in Table 2.5.4–2.

The storage vessel and the discharge vessel are arranged within the reactor pressure vessel. The storage vessels are located above the thermal cover shield through the side reflector; discharge vessels are integrated to the metallic core support structure. The transport line and the transport gas feedback line are laid in the annular gap between the core vessel and the side reflector. The three fitting blocks are connected to the three fittings in the reactor pressure tank bottom. The transport blower including the flow measurement and dust separator are housed in the fuel element discharge room.

Table 2.5.4-2: Design data for small pebble shutdown system

Small Pebble Shutdown Elements

Diameter	approximately 10 mm
Material	B_4C in Graphite matrix (10 volume percent B_4C-fraction)
Number of the small pebble shutdown system per side reflector slot	2.4×10^5
Collapsing time of the total length	approximately 1 minute

Pneumatic Suction Transport Unit

The technical layout of the transport blower is undertaken for the transport in a single unit and for the following system states:

– Cold start-up for normal operation	
Primary circuit pressure:	39 to 60 bar
Cold gas temperature:	50 to 250 °C
– Maintenance	
Primary circuit pressure:	1 bar
Cold gas temperature:	50° C

Operative Functioning During Intended Design Operation and During Failures

The shutdown of the reactor by means of the small pebble shutdown system takes place through the falling in of the small pebble shutdown elements stored in the storage vessels into the slots located in the side reflector and the resultant neutron absorption. To initiate a shutdown, the lifting magnets of the vessel lock of all storage vessels are simultaneously made powerless. The locks open under gravity and allow the shutdown elements to emerge from the storage vessels and fall into the reactor holes. It is possible, due to the special design configuration of the vessel locks as backup locks that even partial quantities could be emitted from the storage vessels.

The return transport of the shutdown elements from the reflector holes into the storage vessel takes place by means of a pneumatic suction transport. Primary coolant at cold gas temperature is used as transport medium. Only the shutdown elements of a small pebble shutdown unit are transported every time.

The suction nozzle on the transport tube within the exhaust vessel is designed in a way that with opened transport bypass fittings, the gas velocity at the inlet into the suction nozzle is smaller than the sinking speed of the shutdown elements. If the transport gas bypass fittings are closed, then the gas velocity at the entrance to the suction nozzle rises to a value above the sinking speed of the shutdown elements and the transport sequence begins. The shutdown elements reach into the storage vessel through the transport tube and the cyclone separator. The transport process can be interrupted at any time by opening the transport gas bypass fittings. However the discharge of the delivery pipe is empty.

In order to ensure a constant delivery rate in the individual system states, at which the density of the transport gas and thereby also the sinking and the delivery rate changes, a speed regulation of the transport blower has been provided for.

According to the Safety Report and the functional description / 2.5.4–6/, the lifting magnets of the backup lock are connected to the storage vessels with the uninterrupted emergency power supply.

A fill level measurement of the small pebble shutdown elements has been provided in the storage vessels as safety regulation instrumentation. Furthermore, a gas throughput measurement before the transport blower, feedback positioner and a speed control are said to be installed as operational instrumentation. If required, the control of the small pebble shutdown system is executed manually /2.5.4–6/. The automation of the small pebble shutdown system is restricted to an operational control for supporting the operators during the transport operation and the prevention of the transport operation in case of a quick reactor shutdown and as a response to the reflector rod feed run-in limitation /2.5.4–7/.

According to the safety concept described in the Safety Report, all devices in the reactor pressure tank, which are required for shutdown and long term sub-criticality, are laid out based on the stresses resulting from external impacts.

According to the Safety Report, the storage vessels including the backup lock are accessible after the opening of the reactor pressure vessels for maintenance even for loaded core. The devices of the pneumatic transport plant arranged in the fuel element discharge room are also accessible during the reactor operation. The discharge vessels are however not accessible.

Tests and Prototype Tests

Drop tests were carried out with shutdown elements and wear measurements on shutdown elements in a pneumatic transport system for testing the fundamental suitability of the small pebble shutdown elements /2.5.4–6/.

The applicant has provided for other suitability tests /2.5.4–8/. Provided for, in detail, are tests of components and subassemblies in component and subassembly tests under closed-reactor conditions and tests on an experimental loop with all components of a shutdown device. The suitability test of components and sub-assemblies extends to:

- Storage vessel including backup lock,
- Lifting magnet for backup lock with limit position indications,
- Fill level indicator of the small pebble absorber in the storage vessel,
- Fittings, and
- Flow characteristic of the small pebble absorber in slot of the side reflector on return transport.

In the suitability test for an experimental loop with all components of a shutdown unit, the investigations named below are supposed to maintain design specifications and demonstrated functional safety:

- Effect of pressure and atmosphere on the characteristic of the small pebble absorber on return (transport characteristic, metering, interruption in transport, wear, separation of the small pebble absorber in cyclone of the storage vessel),

- Smooth functioning of the mechanical and electrical equipment (interaction of all components).

- Checking of the design data (opening and closing time of backup lock, incidence time, input of small sub-quantities, electrical characteristics, return period, metering of the return quantity),

- Proven service life corresponding to the design with rechecking of the design data (shutdowns, actuations on load change, actuations on periodical tests, return),

- Tests of assembly and disassembly procedures including the tools required for them,

- Inspection subsequent to the tests.

As an important criterion of quality, we have based the BMI Safety Criteria on the safety criterion 5.3. In our opinion the KTA rule 3103 "Shutdown System of Light-water Reactors" cannot be applied. Subaspects, such as the generally cited design specifications based on safety regulation, can however be the basis for assessment. This also applied to the RSK guidelines, Section 3.1.2 "Shutdown system."

The proposed planning for the small pebble shutdown system is the design of a new kind of shutdown device. The functional safety of the devices is to be proven in conformance with the state of the art of science and technology for shutdown devices, which have to meet the technical regulation requirements. The fundamental basis for such a proof, for a nonexistent operational test, is a suitability test on a prototype of the new conceptualized shutdown device.

For the primary safety regulation function of a shutdown device, the reliable release of the fall path of the small pebble shutdown elements from the storage vessel in the long hole boring is of consequence. The release of the fall path takes place by opening of the vessel lock. The design is very simple and has a minimum of moving parts. The important requirements for the lock are:

- Reliable opening,
- Avoiding bridge formation on outflow,
- Release of sub-quantity without damage to small pebble shutdown elements on closing the lock.

We are of the opinion that these requirements can be met with the proposed design principle and the functional safety of the lock can be proved in conjunction with the suitability tests.

The other components of the storage vessel, such as the lifting magnet for the backup lock, its position indication as well as the fill level indication of the small pebble absorber in the storage vessel, have to fulfill primarily a function with a regard to the plant availability for operation. We also consider these components as realizable and the functional safety as provable. Furthermore, these components are accessible for maintenance jobs.

The pneumatic suction transport plant also has to fulfill operational functions in view of the plant availability. The important component of this plant is the discharge vessel, which is not accessible in case of any maintenance jobs due to its integration in the core support structure. A special importance is therefore to be attached to the functional safety of this component. We consider the proposed design of the discharge vessel to be impenetrable. We consider the proof of functional safety plausible. However, we start from the premise that within the framework of the suitability testing, tests are conducted on a true-to-scale experimental loop, on which even the long holes are made in the side reflectors under practical conditions and that difficult conditions are assumed, to cover stresses resulting from a long term operation. For example, transportation is to be understood as with allowance for graphite dust in varying quantities, to simulate wear or loosening of abrasion discharges.

The system design is functional, compatible to the tasks. The proposed arrangement offers short transport paths and accessibility of all active components – the discharge vessel is a passive component – for maintenance jobs. The transport gas blower and the fittings blocks are also accessible during plant operation. The design of the pneumatic suction transport system covers the expected operating states in view of the primary circuit pressures and cold gas temperatures sufficiently. A constant transport speed of the small pebble shutdown elements can be ensured even in case of delivery rate related to the operating state, by means of speed control of the transport blower.

There are no special requirements for the energy supply of the small pebble shutdown system from safety regulation view point, as the lifting magnets for the vessel locks are designed to open without power. Only for reasons of plant availability for operation, the lifting magnets are connected to the uninterrupted emergency power supply.

We consider the proposed instrumentation to be adequate for monitoring the functioning of the small pebble shutdown system. In new types of processes, such as capacitive fill level measurement of small pebble shutdown elements in the storage vessel, tests are provided for proving suitability.

The control of small pebble shutdown system is supposed to take place through manual intervention. This is allowable on the basis of the results of the failure analysis. We consider the use of an operational control for supporting transport operation of the pneumatic suction transport plant as acceptable, because the transport process is interrupted if safety engineering falls short of the minimum rod height of the reflector rods.

The safety engineering functioning of the small pebble shutdown system is ensured even on manifestation of such an effect, by means of the proposed design of all devices required in the reactor pressure tank for the shutdown and maintenance of long term sub-criticality, against the stresses resulting from external impacts.

Due to the accessibility of all active components, there exists the possibility to carry out service and maintenance work to a large extent.

In summary, we affirm that we consider the proposed new concept of small pebble shutdown system as feasible, proof can be furnished and we consider this system suitable to fulfilled the tasks resulting from the shutdown concept of the HTR module.

2.5.5 Physical Design of the Reactor

2.5.5.1 Boundary Conditions of the Reactor Characteristics

The characteristics of the reactor core during normal operation and during failures are determined by the design of the fuel elements and of the reactor core as well as the design and operation of the systems in the region on the reactor core. To elucidate this matter, we have presented below a comprehensive reactor core design and the relations to the bounding systems, which are vital for the reactor physical design.

Reactor Core Design

The reactor core consists of a loose charge of about 360,000 spherical fuel elements in a state of equilibrium. In the first core and during the start-up phase, a given portion of the fuel elements is replaced by moderator elements and absorber elements. The cylindrical reactor core with a diameter of about 3 m and a mean height of about 9.4 m is conceived as a single zone core, in which the fuel elements are supposed to obtain a target burn-up of 80,000 MWd/mg U for 15 core pass through in 1,007 days. A graphite reflector with a wall thickness of 1 m surrounds the reactor core in radial direction. On the outside, this reflector is 25 cm thick with approximately 5 volume percent of boron carbide or an equivalent neutron poison. On the inside of the radial reflector, there are holes to accept six reflector rods and 18 holes for accepting small absorber spheres (KLAK).

The flow lines of the pebble bed are determined on the floor reflector by the 30 degree slope of the cone shaped floor reflector. The floor reflector is also provided with cold gas holes, like the roof reflector, through which the cold gas flows in a downward direction for the reactor core. The supply of the cold gas takes place through 72 holes in the radial reflector, which transport the cold gas in axial core direction to the cold gas collection chamber above the roof reflector.

Operative Data of the Reactor Core in Normal Operation

The rated power of the reactor is 200 MW, corresponding to a mean power density in the reactor of 3 W/cm^3. The cooling of the reactor and heat transfer to secondary circuit are done with Helium, which is pumped with a 6 MPa (60 bar) pressure in the primary circuit with a flow rate of about 85 kg/s. The bypass component of the primary coolant flow rate during passage through the reactor core is indicated as 5%.

The moderator, fuel and reflector temperatures determining the reactor state are defined by the reactor power, the coolant flow rate and the cold gas temperature, which is 250°C on entry to the reactor pressure tank. The enthalpy rise of the coolant is 440°C and the radially determined Helium temperature on the exit from reactor core is about 700°C.

The following limitations are provided for the part-load ranges.

	Part-load range
Primary core:	90% –100% of the nominal reactor power
Equilibrium core:	50% –100% of the nominal reactor power

For part-load operation, the coolant flow rate is reduced, so that the enthalpy rise is more or less retained. The cold gas temperature is in the range from 200°C to 250°C, the hot gas temperature, depending upon the load alteration speed, is between 600 and 700°C.

There are fewer temperature changes in the graphitic inclusions and the mean temperature of the reactor core due to this control concept. We have not considered the possibility addressed by the applicant, of operating the individual module at 20 to 50% even in part-load range, because in particular the proof for shutdown safety is provided only for the range 50 to 100% of the rated power.

After a quick shutdown, a hot initiation within about an hour is possible for an equilibrium core; for longer shutdown periods, the decay of Xenon poisoning must be awaited, before a repeat process can be carried out. In case of a long term shutdown, afterheat discharge is done through the main heat transfer system. In case of failure of the main heat transfer system, the surface cooler takes over the afterheat dissipation besides the reactor pressure vessel. In every case, the operational heat dissipation is interrupted in case of a quick shutdown, so that the temperatures in the reactor rise, apart from a short-term temperature equalization of the fuel elements, and a reactivity gain due to cooling is minimal.

Limitation of the Operating Parameters of the Reactor Core

Transient processes and failures are determined in their progress totally by the almost homogenous distribution of the fuel in the form of coated particles in the moderator graphite, through the low power density in comparison with the light-water reactors and due to the neutron-physically inert coolant Helium. For maintaining reliable reactor states for transients and failures, limitations and reactor protection reactions are provided for.

- Limiting the reactor performance to 105% of the reactor power /U 2.5.5–1/.

- Limiting the reflector rod depth of immersion on starting /U 2.5.5–3/,
- Limiting the reflector rod depth of immersion during power operation /U 2.5.4–7, U 2.5.5–2/,
- Locking the KLAK transport in case of too low reflector rod position /4/,
- Limitation of the hot gas temperature within the range ≤ 750°C and the cold gas temperature within the range ≤ 280°C based on threshold values for the reactor safety assembly /U1/.
- Reactor safety-nucleation on exceeding the threshold value by 120% for the thermal-corrected neutron flow /U1/,
- Reactor safety nucleation for a reactor period of ≤ 20 seconds /U 1/.

There are further reactor safety assembly reactions, which lead to shutdown in case of higher humidity in the primary circuit, on too large a drop in the neutron flow and too great a neutron flow without release of the power density.

The values of core design collated here, the provided operating states and the determination through limiting and reactor safety assembly-threshold values basically characterize the reactor-physics behavior, which we shall assess below.

2.5.5.2 Evaluation Basis

We consider the proposed design of the reactor core based on the following basis of evaluation:

Section		Evaluation basis
2.5.5.3	Calculation method	KTA 3104 /L 43/, KTA 3101.2 /L 48/
2.5.5.4	Power density distribution	BMI Criterion 3.1/L 6/, Design of Safety Criteria for Power Generation Plants with Gas-cooled High Temperature Reactors /L 7/, RSK Guidelines for Pressure Water Reactors, Section 3.1.1 /L 10/
2.5.5.5	Shutdown safety	BMI Criterion 5.3 /L 6/, Design of Safety Criteria for Power Generation Plants with Gas-cooled High Temperature Reactors/L 7/, RSK Guidelines for Pressure Water Reactors, Section 3.1.2 /L 10/, KTA rule 3103/ L 45/
2.5.5.6	Reactivity coefficient	BMI-Criterion 3.2 /L 6/, design of safety criteria... /L 7/, DIN 25 405 /L 35/, RSK guidelines, Section 3.2 /L 10/
2.5.5.7	Long-term stability	BMI Criterion 3.1 /L 6/, Design of Safety Criteria... /L 7/
2.5.5.8	After-fission heat output	ANS-rule for After-fission Heat Output /L27/, DIN 25 485 /14/
2.5.5.9	Control rod shutdown program	BMI Criteria 5.3, 3.1 /L /, KTA Rule 3103 /L 45/

Based on the main focus, we have considered the underlying HR-specific Guidelines Outline /L 7/ and the BMI Criteria /L 6/. We have applied the above-mentioned rules as applicable to corresponding light-water reactors or as an item list. As an item we have further used the keywords mentioned in the Rulings 13 /L 72/ and the specific design requirements, which are indicated in the summarizing sheet " The Technology of Nuclear Reactor Safety" for reactor cores with fixed moderator /L 96/.

The observance of safety objectives, such as limitation due to spillage of radioactive materials from the fuel elements, is considered in the chapter on failures. Boundary conditions of failure analyses are predominantly assessed in the following chapters, besides the shutdown safety.

In any case the test points and requirements derived from the evaluation principles have precedence over the assessment made in the following sections.

2.5.5.3 Computation Method

Computational Sequence for the Reactor

For computation of neutron-physical characteristic quantities like multiplication factors and power density fission, the neutron transport equation is to be solved. Computation programs are also used in solutions wherein the reactor under consideration is entered as a geometric model and the neutron physical characteristics (diffusion, absorption and fission) of the individual reactor zones as effective cross-section

The applicant has installed his programming system ZIRKUS /U 2.5.5–4, U 2.5.5–5/ for computation for HTR-Modules reactors, in which different program modules are combined and also considers the sphere pouring-in flow in the reactor, the reactor core burn-ups, the fresh fuel elements supply and removal of burnt fuel elements. For geometric model display, the reactor core is subdivided into concentric stream tubes and axial sections /U 2.5.5–4/, in which the neutron physical characteristics of the available mixture of fuel elements and absorber elements from different burn-ups and moderator elements are taken to be as a constant in each case. The activation cross sections, which display the neutron physical characteristics, are determined with the MICROX program /U 2.5.5–6/; the neutron transport equation for diffusion approximation is solved as per the rough procedure within the ZIRKUS program system.

In this connection, effective diffusion constants depending on the direction are installed as per the Scherer and Gerwin method /L 97/ in the empty space above the sphere fill. The empty spaces in the sphere fill are accounted for by a calculation of the diffusion constants as per Behrens /L 98/, in which case the formula is entered as per /L 99/. The fuel materials and graphite temperature in the reactor core are determined from the computed power density distribution, which can be used as input values for spectral computation with MICROX.

The DIFGEN rough diffusion program and the Monte-Carlo-program MOCA / U 2.3.5–7/ are employed in addition to the ZIRKUS program system, specifically for computation of the absorber effectiveness. The absorber fields of the control elements and KLAK-columns which were determined by transport computations are displayed through activation cross sections or the corresponding boundary conditions in case of DIFGEN input.

Activation cross sections are determined, as a function of the stipulated operating parameters like fuel material temperature, moderator temperature, reflector temperature and Xenon density, from the steady-state computations with ZIRKUS, which together with the neutron kinetics parameters and the thermodynamics are the basis for the dynamic programs ZKIND and RZKIND (refer to section 5.2). With these programs *e.g.*, the reactivity incidents are computed.

Crosscheck for the Computation System

The applicant refers to the re-computations of the critical experiments for crosschecking (verifications and validation) the computation method. For these experiments the reactivity level of critical assemblies is computed within ± 0.5% $\Delta\sigma$. Comparative computations between the MOCA and DIFGEN programs present a consistency, within a narrow range, between the relative effectiveness curves for the control elements and the KLAK-columns. Further re-computations concern the effectiveness of control elements of the AVR reactor, which is produced within 1% tolerance, and power reactivity coefficients of the AVR reactor, which is computed at approximately 10% lower than the measured one. Measurements of reaction rates for the lateral sections through critical assemblies are representative of the neutron flow and with that of the power density distribution. Recomputations of measured reaction rates produce a narrow range tolerance at the core center and a tolerance of approximately 12% at the transition from the core to the reflector.

Computed characteristic quantities of the HTR-Module-Reactors are compared with the results as obtained with the ZIRKUS program system, for crosschecking of the programs ZKIND and RZKIND in case of steady-state computations. Further for the crosschecking of the recomputations of transient processes in the AVR Reactor, are considered the reactivity changes based on the control element operations, blower throughput changes and Xenon transients.

Based on the above-mentioned evaluation principles, we have verified whether

- The ZIRKUS program system is suitable for computing the physical characteristic quantities of pebble-bed reactors,
- The ZIRKUS program system is crosschecked (verification and validation), accordingly errors can be minimized while computing the reactor physical characteristic quantities,
- The model display of the HTR-Module-Reactor core was undertaken properly.

The MICROX program integrated in the ZIRKUS system for computing the activation cross sections and a rudimentary method for solving the neutron transport equation are suitable methods for pebble-bed reactors computations. Especially the mesomerism treatment in MICROX takes into account the fuel material structure of coated particles, integrated in the spherical fuel elements, which exist even during static discharge. We have not considered the subsequent modules in the ZIRKUS system and the evaluation, the data transfer and the burn-ups including rotary flow. We conclude from the concurrence of the details given by the applicant with that of the results obtained independent of the computations that the ZIRKUS program system is absolutely suitable for computing of physical characteristic quantities of pebble-bed reactors.

The recomputations of experimentally determined characteristic quantities provided by the applicant /U 2.5.5–7/ prove that the errors are so miniscule that the characteristic values concerning technical safety regulations can be determined with sufficient accuracy. If the reactor design requires lower error variation, additional measuring techniques are provided to the applicant for crosschecking, *e.g.,* in case of position limiting of the control rod as a demonstration of the safety of the shutdown during the running-in phase. The isolation failures are indicated for each case in the following sections.

The geometric models of the reactor in the ZIRKUS program system, RZKIND and ZKIND are accurately displayed as per our test. The areas to be considered, including the entire reflector, are completely covered. A subdivision of the reactor core into eight spectral zones and 48 burn-ups zones for ZIRKUS program system /U 2.5.5–4/ is adequate for considering the axial profile of the power density distribution, the almost concentric running of the rotary flow in the reactors and the burn-ups differences in the fuel elements in different core levels. The neutron physical activation cross sections of the individual reactor and reflector zones, which are condensed into four energy groups, are subdivided to the required extent for HTR-computations with U 235 fuel material implant.

The quasi-steady-state computations with the ZIRKUS program system and the dynamic programs RZKIND and ZKIND based on it are suitable proofs concerning technical safety regulations because of the proper approach in solving the problem, the crosschecking and the display model of HTR-Module-reactor core.

We depend on our tests computations, undertaken with the V.S.O.P. /L 100/ program system of the Institute for Reactor Development at the Jülich Nuclear Research Plant. With V.S.O.P. /L 100/, using a proprietary data library, one can calculate the activation cross section with the GAK and THERMOS programs, where the mesomerism treatment takes place with ZUT-/DGL. For computing the multiplication factor and the power density distribution, the well-established CITATION diffusion program is used. The fuel material and moderator temperatures are computed with THERMIX.

The correlation of the independently determined results with that of the applicant is good. The tolerances for reactivity equivalents and power density excesses are in a narrow range.

Beside the recomputation with the V.S.O.P. program system, we have consulted publications on testing, especially the ones which have been published during the Annual Conferences on Nuclear Technology especially that on AVR and THTR reactors.

2.5.5.4 Power Density Distribution

2.5.5.4.1 Design of the Power Density Distribution

Due to the neutron leakage at the core edge the neutron flow and thus the power density decrease towards the core edge. This effect leads to a power density distribution that is characterized by power density profiles and power density form factors. With the HTR module reactor core, the axial and radial profiles have an approximately cosinus-shaped profile. The maximum for the primary core as well as for the equilibrium condition is moved to a higher position in axial core direction the power density form factor which defines the surplus compared to the average power density (3.0 W/cm^2) is approximately 1.8. This value, which is results from a zone-related homogenized illustration, is to be superimposed by the surplus in the fresh fuel element. For the intended design with a 15fold flow-through through the surplus in a fresh fuel element is about 1.4.

The nuclear data sets layout resulting from the power density distributions are considered

- In the thermodynamic operation and failure analyses in the form of power and neutron flow distributions,
- In the shielding in the form of neutron flow distributions, and
- In the material capacity in the form of neutron fluence distribution /U 2.5.5–4/.

The power density distribution that results form normal operation is used as an input parameter in the thermodynamic design. There, the safety-technological parameters such as fuel temperature or temperature of metallic RDB installations are considered. We have examined here whether the statements by the applicant are applicable and whether the

Boundary conditions

- Thermodynamic design of power density distribution,
- Neutron- and gamma ray field of the reactor core as source for the shielding concept,
- Neutron ray field of the reactor core as basis of the neutron fluence values for the BE-design and material load factor analysis,

are capable of bearing the load.

We have used the computation results, as determined with the program V.S.O.P. /L 100/, for testing.

Power Density Profiles and Form Factors

The independently performed computations confirmed the details provided by the applicant within a narrow range. For the technical limitations faced during computation with the V.S.O.P program system, of ten cycles per fuel element, we have determined a form factor of 1.87 (Applicant statement: 1.95 /U 2.5.5–4/). For the excess power consumption of the new fuel elements, we have determined the maximum power density as 1.42 (applicant statement: 1.4 /U 1/). We therefore consider the design value for power density distribution under equilibrium state of the reactor core as accurate.

Neutron Fluences of Fuel Elements

We determine the value of the neutron fluence for intended burn ups of 80,000 MWd/kg U at $2.13 \times 10^{21}/cm^2$ (neutron energy = 0.1 MeV). This value confirms the details given by the applicant $2.1 \times 10^{21}/cm^2$ /U 2.5.5–4/.

Neutron Fluence at the Core Boundary

The neutron fluence values at the core boundary display the original values for the material load capacity analysis of the metallic components reflector in the reactor pressure tank (RDB) and that of the reactor pressure tank. Compared to the values given by the applicant, we have determined following maximum values of the neutron fluence:

Location	Applicant /17/	Our calculation
Top reflector	$0.13 \times 10^{21}/cm^2 \times$ year	$0.13 \times 10^{21}/cm^2 \times$ year
Bottom reflector	$0.19 \times 10^{21}/cm^2 \times$ year	$0.19 \times 10^{21}/cm^2 \times$ year
Side reflector	$0.56 \times 10^{21}/cm^2 \times$ year	$0.58 \times 10^{21}/cm^2 \times$ year

Thereby, we consider the applicant's details as confirmed.

In effect, we have determined based on our calculations that the core design details at the state of equilibrium and the original values resulting thereof for the fluence computation are accurate.

2.5.5.4.2 Observing the Admissible Power Density Values

The failures and faults can affect the power density distribution for the steady-state operative condition, which change the power density profile and thereof the power density form factor. The following factors fall under this category:

- Inaccurate movement of the control rod,
- Inaccurate control of a KLAK-column,
- Full level variation of the sphere fill of the reactor core,
- Failure of the BE-supply or BE-recirculation device.

Operative impacts are added to that of the power density distribution, which especially act upon the axial profile of the power density distribution as a result of the burn-ups and recirculation of the sphere fill during the opening phase and which distorts the power density distribution on changing of the load.

Moreover, the applicant explains that the power density falls by approximately 25% in the neighborhood of this rod which has mistakenly moved to the lowermost position, while it increases for the rest of core, as the controls maintain the total output constant /U 2.5.5–8/. The relative rise of the power density is at the maximum at approximately 9% of the core boundary, diametrical to the wrongly moving control rod; at the core axis it is approximately 5%, whereby the maximum power shifts somewhat radially.

If in this situation there is a failure of pressure release then the maximum BE-temperature is not appreciably impacted due to the displacement of the maximum power density to the core boundary and through the core area on dropping of temperature /U 2.5.5–f8/.

An offset of the debris cone of the sphere fill with respect to the core axis is to be disregarded because of the design of the BE-feed pipe in the top reflector. Therefore it is unnecessary to consider the errors in the off-centered debris cone for thermo-hydraulic core design /U 2.5.5–8/.

An inaccurate filling up of the KLAK-column causes the reactor power to drop and in case of rapid filling the reactor is switched off.

In case of slow filling up of a KLAK column and complete controlling with the control rods, the rapid switch-off measure is not effected. The resulting azimuthal unbalanced load of the power density distribution is detected by

- An anomalous reflector bar position,
- The display “backup locking tank open”
- Azimuthal unbalanced load monitoring

/U 1/.

The rupture of the BE- feed pipe within the reactor pressure tank (RDB) could lead to the case that the BE is not conveyed into the reactor core, but depending on the method of rupture it could fall into the bottom area of the RDB or in the annular space between the core vessel and core vessel components. Here as long as the rupture is not detected for a longer period of time (few days), the fill level of the reactor core would fall and the maximum power density would rise, hence the resulting condition for the pressure release under presumed failure would be impaired /U 2.5.5–9/. The fill level control of the sphere fill takes place with multiple BE-counters, not only in case of BE- discharge but also in case of BE-supply /V 2.5.5–10/. A delivery control of the BE is provided within the RDB. Above the top reflector in the metallic portion of the core vessel components, a counter is installed, which frees the delivering of a BE only when the preceding BE has passed through the counting device. The design of the device for delivery control is such that a safe tracking of the fuel elements up to the core is possible, and that in case of failure of the control operation it leads to an automatic adjustment of the BE /U 2.5.5-9/.

In case of failure of the BE re-circulation device, the BE unloading device and the supply of fresh BE, the applicant provides for a drifting operation, for which among others the falling of the fill level of the sphere fill or the rise in the super-elevated power density in the axial core direction is considered as allowed /U 2.5.5–11/. The maximum fuel material temperature was *e.g.*, 10 K above the design temperature in case of a pressure release failure during this drifting, which is not problematic as per the applicant's opinion /U 2.5.5–11/.

On the other hand in case of a failure in supply of fresh BE, the daily requirement of approximately 360 fresh BE is supplied in a batch without mixing with partly burnt-up BE, leads to an excess temperature of less than 1 K as per the estimation of the applicant. Based on the assumption that fresh BE has only fresh BE in its immediate vicinity, as a result of unfavorable statistics the sphere fill of this BE is less than 1.5 K /U 2.5.5–11/.

Here we are considering the faults and failures (occurrences), which can lead to a rise in the maximum power density in case of anomalous operative conditions. This consideration completes the failure inspection of Chapter 5. Operative impacts on the power density distribution like displacement of the axial power density profile in the opening phase or impact of a load reversal operation are assessed in Section 2.5.5.7 (long-term stability).

Considering these delimitations we have verified on the basis of the above-mentioned assessment principles, whether

- The failures and faults are described properly,
- The allowed power density values have been followed,
- The power density distribution accepted as output condition for failures can be followed.

Together with the failures examined in Section 5, the incidents referred to here are complete in view of the power density rise to be considered. We are following the description of the applicant that as a result of the structural design of the BE supply in the top reflector a transfer of the debris cone and an azimuthal rise in the power density resulting from that is not to be taken into consideration /U 2.5.5–2/. Due to the measures specified for the /U 2.5.5–9, U 2.5.5–10/ fill level control, in our view it is not to be assumed that the fill level of the reactor is decreasing unnoticeably and consequently causing a power density rise and temperature rise of the fuel elements associated with it. Statistical fluctuations in the arrangement of neighboring BE with different fuel condition are taken into consideration by the applicant. Temperature rise of the fuels of a few K results from these localized power density rises, which in our opinion are accurate.

Temperature rise of the fuel and the RDB components occurs in case of the incidents assessed here, which lead to the anomalous operating conditions. These are determined clearly in Section 5.2. Design limits of the fuel element and the RDB components are not completely achieved.

The closest approximation to the admissible fuel material temperature is calculated for the coolant loss failure (pressure relief failure, see Section 5.4.1.1). In the analyses conducted in this respect, the most unfavorable operating condition the power density distribution is assumed as 105% of the nominal load of the equilibrium core. Depending on our own calculations, we can confirm the axial and radial power density profile and form factors. The assumption of an error of 10% (2σ value) for the power density distribution in radial or axial maximum is appropriate according to the experimental safeguarding /U 2.5.5.–7/ and the conformance with the calculations performed by us.

Failures or anomalous operation conditions that result in an increase of the power density maximum are not included in the basic conditions of a pressure release failure condition. In case of failures such as control element failure, KLAK miss-triggering, extended operation with a increase of the axial power density maximum, the reactor power has to be decreased in a way that even with the changed power density distribution the fuel element temperatures of 1620°C can be maintained.

The SE position control, the KLAK storage vessel fill status control, the storage indicator of the KLAK storage vessels and through the off-load control with the aid of 3 × 4 measuring positions of the neutron flow instrumentation (see Section 2.13.3), make an adequate control of the power density distribution possible. The described instrumentations are adequate to build up a protective limitation according to KTA regulation 3501 /U 2.5.5.–12).

2.5.5.5 Shutdown Safety

2.5.5.5.1 Reactivity Balances

The safety shutdown is basically determined through the functional safety of the components of the shutdown system (reflector rods in the radial reflector combined with the blower shutdown, refer to Section 2.5.4.2, and KLAK columns for boring in the radial reflector, (refer to Section 2.5.4.3)and by their effectiveness in the neutron balance of the reactor. The neutron balance can be described based on the reactivity, which is once again associated with the effective multiplication factor:

$$= (1 - \frac{1}{k_{eff}}) \times 100 \lfloor \% \rfloor$$

In reactivity balances, the to-be-bound reactivity equivalents $\Delta\sigma$ in connection with a shutdown are compared with the effectiveness of the switch- off system. The resulting shutdown reactivity must be below the limits specified for demonstrating the shutdown safety. Following reactivity equivalents are taken into account by the applicant /U 1, U 2.5.5–14, U 2.5.4–1/:

- Excess reactivity for ruling out the non-stationary Xenon during load alternation operation,
- Reactivity fluctuations as a result of reloading strategy during the start-up phase and in the equilibrium core,
- Reactivity recovery as a result of load disadvantages or in case of a temperature drop of the reactor core,
- Reactivity recovery through isotope decay, especially through the decay of the neutron-absorbing isotope Xe 135,
- Reactivity supply due to a failure.

The description is related to nominal operating condition with $\sigma = 0$. In the primary core and during the initial operative months an excess reactivity of + 1.2% reduced against the equilibrium condition (with $\sigma = + 1.2\%$) is added, consequently the load alternation operation covers only 90 to 100% of the nominal load for the time being. This excess reactivity is linked likewise to a positive reactivity fluctuation because of the reloading strategy or as the additional reactivity input due to Xenon burnout or Xenon decay during the load alternation operation of reflector rods as well as KLAK columns. The reactivity linkage of approximately 45 hours is provided in case of long-term partial load operation in the equilibrium core from reflector rod system to the /U 2.5.5–15/ KLAK columns. Then the reflector rods move back again to the nominal load required position.

It is ensured that by limiting the operational entry that there is sufficient shutdown reactivity in all required /U 2.5.4–7, U 2.5.5–14/ cases. This limitation is justified on the basis of measuring /U 2.5.4–7, U 2.5.5–14/ results for the respective reactor condition (primary core, start-up phase, equilibrium core) /U 2.5.5–15/.

An interlocking device ensures that while starting from the reactor condition "cold zero load" that the KLAK columns can be successively emptied only when the reflector rods were brought to their normal position (nominal load required position) as /U 2.5.5–3/ rod bank.

The shutdown balances for the primary core while inserting the reflector rods as well as filling up of the KLAK columns is indicated in the /U 2.5.5–14/ document.

Reactivity balance of the thermal shutdown system for the primary core:

– Derivative action for load alternation range 100–90–100% including control reserves	+ 0.2%
– Reactivity compensation in case of partial load (Xenon effect, temperature)	+ 0.6%
– Maximum failure reactivity	+ 0.5%
– Partial load-zero load reactivity (thermal)	+ 0.1%
– Effectiveness of 5 of the 6 reflector rods	–3.4%
– Shutdown reserve	–2.0%

Reactivity balance of the cold shutdown systems for the primary core:

– Derivative action for load alternation range 100–90–100% including control reserve	+ 0.2%	+ 0.2%
– Temperature drop of the core up to 50° C	+ 8.0%	+ 8.0%
– Isotope decay	+ 3.2%	+ 3.2%
– Maximum failure reactivity	–	+ 0.5%
– Derivative action for start-up phase	+ 0.5%	+ 0.5%
– Sub-criticality	+ 0.3%	+ 0.3%
– Effectiveness of 18 KLAK-columns	–13.7%	–
– Effectiveness of 17 KLAK-columns and 6 reflector rods	–	–14.2%
– Shutdown reserve	–1.5%	–1.5%

The shutdown balances for the equilibrium core while using the reflector rods as well as while filling up the KLAK columns are indicated in the /U 1/ safety area and in the /U 2.5.5–14/ document.

Reactivity balance of the thermal shutdown system for the equilibrium core:

– Derivative action for load alternation range 100–50–100% including control reserves	+1.2%
– Reactivity compensation in case of partial load (Xenon effect, temperature)	+0.4%
– Maximum failure reactivity	+0.5%
– Partial load-zero load reactivity (thermal)	+0.1%
– Effectiveness of 5 of the 6 reflector rods	–2.6%
– Shutdown reserve	–0.4%

Reactivity balance of the cold shutdown systems for the equilibrium core:

Derivative action for load alternation range 100–50–100% including control reserve	+0.2%	+0.2%
Temperature drop of the core up to 50° C	+3.0%	+3.0%
Isotope decay	+3.6%	+3.6%
Maximum failure reactivity	–	+0.5%
Sub-criticality	+0.3%	+0.3%
Effectiveness of 18 KLAK-columns	–10.6%	–
Effectiveness of 17 KLAK-columns and 6 Reflector rods	–	–11.0%
Shutdown reserve	–2.5%	–2.4%

The shutdown reserves (shutdown reactivity values) resulting from the reactivity balances are available for covering the /U 1/ calculating uncertainties and /U 2.5.5–14/ tolerances.

The shutdown reserve in the primary core is 1% lower than in case of the equilibrium core. If it is specified at the start-up of the reactor that the reactivity balance of the cold shutdown system does not start, then the requirement of the shutdown system can be cut-down /U 2.5.5–14/:

- Transient reduction of the reactor power and lowering of the hot gas temperature,
- Transient rise of the lowest allowed temperature of the reactor core.

The last mentioned possibility is taken into account, if the charged excess reactivity is not sufficient and the power operation is too limited. Then there is a provision to change the assembly of the reactor core and to increase the lowest tolerable temperature of the reactor core after the shutdown. This can be done through heat input by means of circulation of primary coolant or a short-term power operation /U 2.5.5–14/. If the start-up phase has to be interrupted for a longer period, then the reactor is turned cold sub-critically by supplying absorber elements or neutron absorbing materials /U 1/.

The interruption of the primary coolant throughput, which takes place at every fast shutdown of the reactor, is viewed as an additional possibility for reactor shutdown /U 1/. A sub-criticality of the reactor results due to a gradual rising of the core temperature and due to the negative temperature coefficient of reactivity /U 1/.

For protecting the calculated effectiveness of the shutdown elements, the applicant has performed comparative calculations with the Monte Carlo program MOCA. Few per thousands are the tolerance for the computations using the program DIFGEN not only for the entire bank of the reflector rods but also for the 18 KLAK columns /U 2.5.5–16/.

We have verified the neutron physical aspects of the shutdown safety on the basis of the shutdown concept, as considered in Section 2.5.4.1. As per the above-mentioned assessment principles we have evaluated following test points:

- Integrity of the reactivity balances,
- Proof of adequate shutdown reactivity
- Availability of starting limits for shutdown system, which are preferred for controlling tasks,
- Loading strategy (refer to Section 4.3)

We specify that for the proposed reactivity balances the effects on the reactivity in case of a shutdown or removal or the operational effects on the reactivity adulterant are described comprehensively. The reactivity equivalents relevant for a HTR – reactor with Uranium as fuel – likewise the secondary failure of a reflector rod as well as the failure of a KLAK unit for shutdown, are taken into account.

For the first shutdown system (reflector rods), which serves for short-term failure control, we consider a shutdown reactivity of –0.3% based on the measurements carried out at least half an hour after the occurrence of the failure. For the verification, we have considered the reflector rod bank effectiveness by determining it independent of the V.S.O.P./CITATION. Compared to the applicant's statement, the results are as follows:

	Effectiveness of the reflector rod bank	
	Primary core	Equilibrium core
Document /38/	–4.1%	–3.2%
V.S.O.P./CITATION	–4.5%	–3.4%

We concur with the applicant's statement by including a correction of $\Delta\sigma = 0.2\%$ for the neutron current in the reflector rod boring that is necessary for our calculations. The reactivity equivalents to be compensated are likewise accurate as per our test. A reactivity of + 0.5% supplied from outside covers the failures "water leakage" and "compression of the spherical flow as a result of induced shocks" to be taken in to account.

An adequate shutdown reserve while installing the first shutdown system can be obtained by means of the effectiveness measurement of the reflector rods provided by the applicant and the settings of the starting limit of the reflector rods derived from that.

We add a relative accuracy of ±10% for the calculation of the reactivity equivalents and the calculated effectiveness of shutdown elements – also accepted for the appraisal of THTR /L 77/ – which however, in this case leads to a shutdown reserve that is not satisfactorily demonstrated by calculation.

A reduction in this tolerance is not added for the time being even after the IBS measurements on the THTR. However an adequate shutdown safety is proved with the measuring techniques independent of the additional calculation error.

The beginning of the operational measurements for effectiveness of the reflector rods during the start-up phase and its time interval are to be determined before the start-up.

The reactivity balance considered by the applicant for shutdown effectiveness of the reflector rods does not relate to an unfavorable condition in the power operation. We specify that while starting from the condition "zero load cold" a further unfavorable output condition occurs if the reflector rods are lowered due to the shielding effect of started KLAK columns in its effectiveness. Based on the starting frequency for the maximum entry depth of the reflector rods /U 2.5.5–3/ and a suitable operating sequence of the KLAK columns while starting, an adequate shutdown reserve of the reflector rod can be maintained. In case of the construction planning, reactivity balances are to be created for inserting the reflector rods while starting in the unfavorable operating conditions. The operational measurements should be specified for justifying the starting limitations and the interval of measurements before the start-up.

For the required shutdown reactivity while installing the second shutdown system, which should maintain the reactor continuously in the sub-critical condition in unfavorable circumstances, we add the value –1.0% (determined by calculation) based on above-mentioned rules. Our inspection of the reactivity balances with the program V.S.O.P./CITATION confirms the applicant's statement, how *e.g.*, a comparison to the effectiveness of the 18 KLAK columns is covered.

	Effectiveness of the 18 KLAK columns	
	Primary core	Equilibrium core
Applicant /1.32/	–13.7%	–10.6%
V.S.O.P./CITATION	–13.8%	–10.6%

While including the above-mentioned tolerances for the to-be-compensated reactivity equivalents and for the effectiveness of the shutdown elements, the calculation of the shutdown reactivity of the primary core and the reactor core during a part of the start-up phase is not sufficient. Therefore, during the loading process of the primary core it is to be verified with an appropriate shutdown of reactivity via unfilled KLAK columns that the shutdown safety for the cold shutdown of the primary core is given under failure conditions. This measurement is to be repeated during the start-up phase. The assembly of the reactor core should be altered in case of inadequate shutdown safety.

The applicant has given an additional possibility of maintaining the reactor by means of coolant circulation or through an intermittent power operation at an increased temperature level and consequently sufficient sub-criticality, after shutdown. We consider this possibility as an essential safety potential of the HTR concept described here.

We consider a change in the primary core assembly, which the applicant provides on operative grounds in case of insufficient surplus reactivity /U 2.5.5–14/, as appropriate, and also due to the above-mentioned reasons for safe shutdown. As a result of the consistency of the applicant's statement for shutdown safety /U 1, U 2.5.5–14, U 2.5.5–16/, which were partly determined with different programs, and with the values selected by us, we seem to have crosschecked the computation properly. But as the safety technical measures available at present are not sufficient for reducing the tolerances, the above-mentioned measures are necessary.

For the KLAK system, which must alone be able to shutdown the reactor from each operating condition and maintain the sub-critical status, we decide on a necessary shutdown reactivity of −1% (determined by calculation). This shutdown reactivity can be maintained by considering our above-mentioned measures, as the shutdown reserve of the KLAK system is almost the same by itself or in combination with the first shutdown system.

The supply of absorber elements or absorbers, which the applicant provides, appears positive to us; in case the system operation is interrupted for a longer period /U 1, U 2.5.5–14/, the sub-criticality can be increased. But we cannot accept the integration of this supply of absorber elements or absorbers in a shutdown system for reaching and maintaining the "cold sub-critical status". However we consider the supply of absorber elements and absorbers necessary to set an adequate sub-criticality for measures and incidences like repairing of the shutdown system, opening the RDB or failure of KLAK column.

We view the reactor shutdown due to the exclusive shutdown of the blower, the temperature rise in the reactor and then the ensuing interruption in the nuclear power generation as the essential safety potential of the HTR-Module. However the described operating method cannot be applied as an independent shutdown measure for a commercially used reactor.

The reactor core operation, considered by the applicant, for a wrongly running control rod or an inaccurately filled-up /U1, U 2.5.5–8/ KLAK column can be performed only for a limited time on grounds of shutdown safety. Depending on the extent of the fault and the repair possibilities, an operation with reduced reactor power or a shutdown of the reactor is necessary.

A fill level limitation of the KLAK columns with the help of the fill level indication of the KLAK storage tank for ensuring the shutdown function is necessary, as the KLAK system is also responsible for operational tasks as a part of the safety system, *e.g.*,

- Filling the KLAK borings for preventing the neutron flow in the lower RDB area,
- Reactivity absorption for long term partial load operation at 50% of the nominal load,
- Reactivity absorption of approximately 0.5% from the reliable bandwidth of the surplus reactivity during the running-in phase.

2.5.5.5.2 Reactivity Ramp, Boundary Conditions of Failure Analysis

The reactivity ramp as a change in the reactivity per time unit is important as negative reactivity ramp for the shutdown process, for which the reactor must be made powerless as fast as possible, and a positive reactivity ramp for failures for which the reactivity supply must be limited per time unit and overall.

The changes in the reactivity as a function of the entry depth of the reflector rods and as a function of the KLAK columns fill level are indicated in document /U 2.5.5–16/. The computations were performed with the MOCA Monte Carlo Program, and are additionally is considered by the applicant as a computational crosscheck of the shutdown safety. The reactivity ramps to be taken into account together with the track speed of the reflector rods and with the draining and filling speed of the KLAK columns are computed /U 2.5.5–16/.

The reactivity as a function of the filling factor of the core element discharge is indicated in the Safety Report /U 1/. The reactivity increases by about 0.19% with the relative increase in the fill level by 1% for the equilibrium core. The reactivity increases by approximately 4% by removing the reactivity absorption with the reflector rods. The induced vibrations of the reactor core result in temporal progress of the filling factor rise along with the ramp and the reactivity supply. The applicant determines a reactivity supply 0.125% for a horizontal seismic speed of 0.5 g (1 g = 9.81 m/second^2) with an excitation time of 6 seconds /U 1/.

The reactivity process as a function of the waterproofing in the equilibrium core is indicated in document /U 2.5–1/. The rise in reactivity of the reactor core is approximately proportionate to that of the waterproofing up to a maximum water quantity of 600 kg in the primary circuit (refer to Section 5.3.3). We determine the maximum reactivity supply to be assumed at 0.4% from document /U 2.5–1/.

The analyses for the reactivity anomalous incidences are evaluated in the Section 5.2. We have verified here, whether the corresponding limiting conditions of the analyses are complete on the basis of the reactor physical core design and operation planning of the system adjoining the reactors. Additionally we have verified, whether the KLAK system is suitable for compensating the most positive reactivity ramp occurring in the determined operation.

Reactivity Ramp in Operation

The highest positive reactivity ramp in the operation is a result of the burn-out of the Xenon poisoning, which occurs while starting up after a shutdown or while starting up from 50% of the nominal power. As per our test, the KLAK system is suitable for compensating this ramp and consequently to shutdown immediately if necessary.

Faulty Running, Faulty Incidence, Ejection of Reflector Rods

We have determined the effectiveness of a rod bank as a function of the entry depth with our computations. The results of the computations confirm the /U 2.5.5–14, U 2.5.5–16/ applicant's statement (also refer to Section 2.5.5.5.1). Therefore, we consider the limiting conditions of incidence analyses in the Section 5.2 as appropriate for taking into account the faulty running of the reflector rods. A starting limit for the rod entry depth can avoid a positive reactivity supply, which is possible while starting up the reactor in case of a faulty incidence of a reflector rod (refer to Section 2.5.5.9). In our opinion the ejection of a reflector rod should not be assumed, as the reflector rods including that of the drive are mounted in the pressurized enclosure. The pressure differences occurring due to the incidence of the pressure release are not sufficient to affect the raising of the reflector rods.

Faulty Emptying of KLAK Columns

We consider on the basis of our computation that the details in the applicant's statement /U 2.5.5–16/ are appropriate for the effectiveness of the KLAK shutdown system. A positive reactivity supply through the KLAK system is possible because of the design of the KLAK delivery blower and the controls, at each case only in case of empty lifting of a KLAK column. The KLAK shutdown system is designed and arranged in the reflector in such way that a systematic emptying of the KLAK columns is not to be assumed in the normal operation and in case of the underlying anomalous incidences.

Geometric Changes of the Core Element Fill and the Reflector Fill

We have examined the following possibilities of a positive reactivity supply:

- Sudden fall of the top reflector,
- Break-down of the sphere discharge pipe, leakage of the core element fill in the area below the RDB,
- Change in the volume/surface ratio of the core element fill due to spontaneous expansion of the radial reflector,
- Vertical displacement of the radial reflector or the core tank opposite the core element fill.

We specify that the above-mentioned possibilities are not to be assumed to be failures, due to the design of the RDB, due to the connection of the reflector in the core tank and due to the design of the supporting structure of the top reflector. The generally considered sudden fall of a graphite block from the top reflector leads to a negligible reactivity supply as a result of the core height of 9 m, of the axial symmetrical flow profile, and the slight axial leakage resulting from that.

We have performed computations for the primary core for the reactivity process as function of the fuel flow-filling factor. Our computations, under consideration of the incorporation of the leakage ratio between primary and equilibrium core, confirm the applicant's statement /1/, which were made for the equilibrium core.

Water Leakage

We have computed the reactivity process as a function of waterproofing in the reactor core respectively for the hot and cold reactor condition for the primary and equilibrium cores. Our computations confirm that a maximum reactivity supply of 0.4% covers the assumed waterproofing in the primary circuit of the equilibrium core. There is negligible reactivity supply even under unfavorable conditions in case of the primary core. In this case, a slight-water-proofing in the core would reduce the reactivity of the reactor.

Temperatures in the Reactor Core and in the Reflector

The reactivity is supplied by decreasing the temperature of the reactor core and by increasing the temperature of the reflector. The neutron physical coupling is determined with the reactivity coefficients (refer to Section 2.5.5.6). We have examined whether power can be released though the Wigner Effect, especially as a result of the positive temperature coefficients in the reflector area. We confirm that the temperatures of the graphite components in the power range are so high that significant heat release input and the reactivity supply is not to be expected in case of an failure causing heating up.

Summary

Our verification of the reactivity ramp has produced the result that the reactivity failures as discussed in the Section 5.2 are complete from the reactor physical point of view. The quantitative verification of the relevant positive reactivity ramps confirms the applicant's statement.

2.5.5.6 Reactivity Coefficient

2.5.5.6.1 Range of Values of Reactivity Coefficients

The reactivity coefficients indicate the dependency of the reactivity from the parameters that impact the neutron balance and the power release of the reactor. Following parameters are mainly taken into account for HTR-type reactors:

- Moderator temperature (temperature of graphite of the core element),
- Fuel temperature,
- Reflector temperature.

The reactivity coefficients change with the operating condition of the reactor. In this case, the burn-up condition of the primary core to the equilibrium core, the temperature level of the reactor, the Xenon concentration and the position of the absorber elements are important.

A back coupling results from the reactivity coefficients, as the power release depending on the operating parameters affects the operating parameters like fuel or moderator temperature. Within this back coupling the reactor design is designated as inherently safe, if the sign of the specific reactivity coefficients are oriented in such a way (*i.e.* negative) that every increase in output is isolated.

The quick-acting fuel temperature coefficient is always negative. This coefficient, which results mainly from the temperature dependency of resonance absorption in the U 238 and Pu_{240} isotope, depends on the burn-up condition and the fuel temperature itself. The value range $-8\times10^{-5}K^{-1}$ to $-4\times10^{-5}K^{-1}$ is indicated for the equilibrium core in case of an average core temperature of 100° C to 900° C. For the primary core, the value range of fuel temperature coefficient is about half as big as a result of displacements in the neutron spectrum and in the neutron leakage /U 1/.

The moderator temperature coefficient is negative in the cold reactor of the equilibrium core, almost zero in the temperature range of the power range, and again negative for higher temperatures. This coefficient resulting mainly from the effect of neutron spectrum on the fission and absorption in case of the Uranium and Plutonium isotopes in the thermal energy range is clearly negative in the primary core. It is characteristic for primary core that the total temperature coefficient is shifted considerably in the negative range in proportion to the comparable core conditions for the equilibrium core. As the fuel is distributed almost homogenously in one part of the moderator, it is therefore ensured that the temperature will affect the reactivity sufficiently fast /U 1/.

The total temperature coefficient of the reactivity (total of the fuel and the moderator coefficient), is always negative as a result of the negative input of the fuel coefficients. The effect of the Xenon concentration, which leads to a positive value for the Xenon maximum for moderator coefficients of the equilibrium core, produces the maximum value of $-4\times10^{-5}K^{-1}$ /U1/ for the total temperature coefficients of the equilibrium core.

The reflector temperature coefficient resulting from the increased back-scattering capacity of the graphite at higher temperature is weakly positive in the entire operative temperature range; the value range is +1 to $+3\times10^{-5}K^{-1}$ for the equilibrium core /U 1/.

We have verified on the basis of the above-mentioned evaluation principles, whether

- The reactivity coefficients to be taken in to account are indicated completely,
- The fundamentals effects on the range of values of the reactivity coefficients are considered,
- The range of values of the indicated reactivity coefficients are accurate,
- The prompt input of the temperature coefficient of the reactivity is sufficiently negative in the determined operation and in case of failures,
- The positive reactivity coefficient is limited in such a way that additional power inputs do not arise or are not taken into consideration in case of failures.

According to our knowledge, the reactivity is predominantly dependent on the operating temperatures, fuel temperature, moderator graphite temperature, and graphite components temperature. Therefore we regard the reactivity coefficient considered by the applicant as complete.

The neutron physical effect on the reactivity coefficients are based on displacement of the neutron energy spectrum, which can mainly arise through temperature, burn-up (transfer from primary core to equilibrium core), Xenon concentration and absorber, like reflector rods and KLAK-columns. The stated effects are displayed and described by the applicant; again we also regard the applicant's statement as complete.

We have performed spot check-type computations for the range of values of the reactivity coefficients for the primary core, which confirm the applicant's statement.

We have determined reactivity equivalents between nominal load and "cold zero load" conditions for our computations, which correspond, within a narrow range, with the applicant's statement /U 1, U 2.5.5–14/. These reactivity equivalents display the reactivity coefficient integrals concerning the temperature. We conclude from its correlation that the reactivity coefficients concerning the details of the range of values are also accurate. The total temperature coefficient composed from the fuel and moderator coefficients is shifted almost by a factor of two in the negative range in the primary core opposite the equilibrium core, although the absolute value of the fuel coefficient is very low. We agree with the applicant that an almost homogenous mixture of the fuel and moderator coefficients ensures that the moderator coefficient in the primary core is integrated with the prompt negative back coupling.

The surplus reactivity affects the neutron energy spectrum with the aid of the Xenon concentration, and as a result, the moderator coefficients. The positive contribution of the moderator coefficient in the equilibrium core is maintained by dimensioning the surplus reactivity of the reactor core, in such a way that a prompt negative reaction in the equilibrium core is maintained through the total temperature coefficient or the power coefficient.

Likewise, the positive temperature coefficient of the reactor does not have any significant effect on the power release. Assumptions based on the applicant's statement and our computation show that a temperature related reactivity supply is always compensated by a decrease in the reactor reactivity due to the comparatively small absolute amount of positive reflector coefficients. A fast reactivity supply just by increasing the temperature of the reflector is not suggested as a result of the reflector dimensions, the cold gas feed in the borings of the /U 2.5.5–4/ reflector and also because of the significant decrease of the Wigner Effect (refer to Section 2.5.3.1).

Reactivity changes following changes in the coolant thickness are not given, as the coolant does not react with Helium.

In summary we determine that the total reactivity coefficient and the power coefficient resulting from it cause only a prompt negative back coupling in the operational and anomalous incident range under consideration. The requirements for the inherent safety of the reactor are maintained.

2.5.5.6.2 Reactivity Coefficients, Boundary Conditions of Anomalous incident Analysis

The reactivity coefficients together with the input of delayed neutron and thermodynamic design values of the reactors determine the integral and local power release of the reactor during anomalous incidents or failures. It is possible by coupling the reactivity to the operating temperatures like moderator and fuel temperatures to produce a reactivity supply and thereby associated increase in output during anomalous incidences or in case of failures. On the other hand, the back-coupling resulting from the negative reactivity coefficients suppresses the power release (refer to Section 2.5.5.6.1).

The inspection of the anomalous incidences is evaluated in Chapter 5. At this point we have examined, whether the determined reactivity coefficients are taken completely into account in the anomalous incidence analysis, and whether the possible range of values of the reactivity coefficients are included completely in the details of the anomalous incidences.

The ZKIND and RZKIND programs /U 2.5.5–5/ installed particularly for the anomalous incidences occurring due to reactivity (refer to Section 5.2) completely take into consideration the relevant couplings affecting the reactivity under operating conditions, in the physical neutron section. Now only temperature decrease in the reactor core has to be considered for a reactivity gain that takes place by indirectly coupling the reactivity coefficients during anomalous incidences. This temperature decrease is stopped by investigating the start-up of the primary circuit blower and decreasing the cold gas temperature (refer to Sections 5.2.4, 5.2.5).

In addition to the reactivity gain by lowering the temperature, we have also considered the possibility of reactivity gain by evaporating the water in the reactor core in case of the anomalous incidence of "water leakage". We have considered this possibility within the reactivity coefficients limits, as the reaction is of importance at the reactor core temperature for the waterproofing. Starting point is the reactor reactivity process, which indicates a sub-critical range at several 100 kg water in the primary core and several tons of water in the equilibrium core. Under this condition, rise in the temperature can lead to water evaporation and reactivity gain. However the results of our verification show that for reaching the presumed starting condition, the safety measures taken for the reactor safety should have failed and the waterproofing in the reactor core should have recorded the values immediately during the starting process, as if the water had trickled through the sphere fill. The last mentioned condition couldn't occur just on the basis of the arrangement of the evaporator beside the reactor, and together with the boundary conditions the considered anomalous incidence process should be regarded as unrealistic. Hence in our opinion during the course of a failure, a significant recovery of reactivity occurring in the phase transfer from water-to-vapor cannot be conceived.

2.5.5.7 Long-term Stability

Temporal and local changes in the Xenon concentration (Xe 135) which affect the neutron flow and consequently also the power and power density distribution due to the high effective cross-section of the Xenon should be particularly considered with respect to the long-term stability of the reactor. Furthermore, local changes in the fission fuel and fission product concentration occur until the equilibrium core condition is reached, which likewise affects the power density distribution. Failures in the BE-feeder, during BE-burn up determination and in the BE-extraction unit can immediately affect the power density distribution. In this case the balance between the fission fuel feed, burn up and circulation is altered, as a result the power density distribution is displaced, particularly in the axial direction of the core.

The applicant explains that the maximum impact of the effect of the Xenon concentration on the power density distribution should be expected during load alternation process while increasing the reactor power from 50% to 100% /U1/. A maximum increase of about 30% is determined from the temporally and locally changing power density distribution in the axial core direction, which occurs within about 5 to 6 hours after starting the reactor power. This increase results from the local re-distribution of the Xenon concentration in the reactor core and the change in the controls of the control elements, which compensates the integral changes in the Xenon concentration.

A study was carried out to demonstrate the stability of power density distribution against Xenon fluctuations /U 2.5.5–19/, in which the determining reactor physical effect levels (diffusion constants, temperature coefficient, and reactivity absorption by Xenon) were changed. It showed that an increase in the diffusion constants or the core height by approximately 50% would be necessary to achieve the sustained Xenon fluctuations. Xenon fluctuations are dampened for the reactor core considered here /U 2.5.5–19/.

The progress of the power density distribution during the opening phase (transfer from primary core to the equilibrium core) is not displayed, as at present only the primary considerations of the applicant are presented here. Subsequently, the applicant professes to limit the power per fuel element to 4 KW in the start-up phase and the power per fuel particle to 250 mW and the surplus reactivity for a load alternation range of maximum 100–5–100% of the nominal power. Additionally the maximum fuel element temperature should be limited to 1,620° C for all assumed anomalous incidences /U 2.5.5–1/.

We have worked out following test points while formulating the above-mentioned evaluation principles:

- Stability of the power density distribution,
- Maintaining acceptable power density values under normal operation,
- Maintaining acceptable power density values as starting conditions for anomalous incidences.

The effects on the power density distribution considered here concerns the long-term stability; these effects are limited to the causes that are due to the design and operation of the reactor core. Failures that affect the power density distribution immediately or later are considered in Section 2.5.5.4.2.

In our opinion it is adequately checked, according to the studies carried out by the applicant /U 2.5.5–19/, that the power density distribution is stable against Xenon fluctuations even in axial core direction. In spite of the relatively bigger core height of approximately 9.4 m the ratio of core diameter to core height is selected at 1:3; as a result the power density distribution is quasi "blocked" and Xenon fluctuations are suppressed in axial direction. Consequently, special instrumentation, control systems or the limitations for detecting and controlling the Xenon fluctuations with drifting un-dampened amplitude are not necessary.

Maintaining Allowable Power Density Values during Normal Operation

The applicant has explained in the Safety Report that the power density is increased maximally during the normal operation and this increase occurs during the load alternation process and while increasing the reactor power from 50% to 100%. We regard the indicated increase as accurate on the basis of the power density distribution

According to our opinion the power density distribution during the start-up phase is controlled in such a way that the allowable fuel temperature can be maintained safely during normal operation.

Due to the design features as a single zone core with relatively higher circulation speed, the indicated bandwidth of the Uranium 235 concentration (refer to Section 2.5.1), and the absorber element design /U 2.5.1–1/ which is yet to be determined, design parameters are given and adequate control parameters are available to keep the power density distribution within the allowable limits during the start-up phase.

We have additionally considered the effect of the rotary flow of sphere fill on the power density distribution. It is demonstrated using the inspection results available in the literature /L 101/ and the applicant's statement /U 2.5.5–18/ that bridge formation with a retention of sphere flow cannot occur for the existing ratio of sphere outlet diameter to the fuel element diameter in the HTR module. A static loosening of sphere fill in the forcing cone/sphere outlet cannot be excluded in our opinion. Such loosening would produce a fill level fluctuation in the range of a few millimeters, as it also occurs as a result of statistical fluctuation in the fill cone of the core fill. These fluctuations of the fill level correspond with the insignificant changes of the fill factor (refer to Section 2.5.5.5.2), which once again affect the fluctuations of the reactivity and consequently a "noise" in the integral reactor power. In this case, the given response of the reactor power is limited to 105% of the nominal power /U 2.5.5–1/, but significant effects on the power density distribution are not to be expected.

Maintaining Allowable Power Density Values as Starting Condition for Anomalous Incidences

The applicant regards the anomalous incidence of pressure release as the anomalous incidence leading to the maximum BE-temperatures, not only for the start-up phase /U 2.5.1–1/ but also during a load alternation operation /U 1/. Quantitative proofs of the BE-temperatures for the pressure release anomalous incidence during this operating condition can be still found. Proofs for maintaining the allowable fuel element temperature of 1620° C are fundamentally possible even for these cases. However reduction in the reactor power and the load ramps are determined in this case if necessary.

2.5.5.8 Decay Afterheat Power

The applicant shows in the Safety Report that he has determined the afterheat on the basis of the DIN 25 485 /L 36/ and the ANS-method /L 27/ HTR-specifications and the ORIGEN computation program. The ANS-method determines the input of directly formed fission products for decay afterheat power; the ORIGEN program determines the input by neutron capture reaction in the fission products and disintegration of the actinides. According to the /U 2.5.5–20/ work report the ORIGEN computation method is verified by comparing the results of the ORIGEN with the results of the ZIRKUS computing method. For this computation, the reactor core is subdivided into six axial zones with different neutron spectrums using parameter variations THERM, RES and FAST and with different powers to rebuild an axial power profile. The applicant has shown the nuclide concentration of some actinides within these ranges.

The applicant estimates the error for computing the afterheat production to be an average 2 sigma-error according to DIN 25 485 for the relevant period of 40 hours.

We are of the opinion that the method used for determining the single input of the decay afterheat power is adequate for the HTR-Module. The proposed study for verifying the ORIGEN computation program for using on the HTR show a good correspondence between single inputs determined with different methods for decay afterheat power.

Our computations for decay afterheat power on the basis of the effective cross-sections of the HTR produce insignificantly higher decay afterheat power. Our computation results show that in the set period of 30 hours, the boundary conditions in the burn-up computations, like for example the power density rise and the connected power of the individual fuel element spheres clearly affect the calculated afterheat amount at about 15 times the core flow. As per the applicant's assumption, we view the existing data uncertainties used for the effective cross-section to be adequately considered, and calculate a 2 sigma-deviation corresponding to a fault of 5.6% in the 30 hours period after the shutdown.

Likewise for a power density distribution occurring particularly due to axial Xenon distribution, we regard the error in the form of an additional input for afterheat production with a double standard deviation as per DIN 25 485 as sufficient.

2.5.5.9 Control Rod Shutdown Program

The always-present surplus reactivity of the reactor core is compensated using the reactivity adulterant control elements and/or KLAK columns during each reactor operation period. As these reactivity adulterants are inserted as shutdown system, besides the operating tasks and the reactivity calibration agents affect the power density distribution as local neutron absorber, it produces the safety regulations of the control rod operating program described below.

While starting processes from the "cold no load" reactor condition the KLAK feed can be released only if the reflector rods bank were drawn to its normal position /U 2.5.5–3/. This is guaranteed by a locking mechanism. An operating process in four groups is provided for the KLAK system, where the elements of a group are divided as evenly as possible around the reactors /U 2.5.5–3/. Three groups consist of five KLAK columns each; one group with three KLAK columns is always controlled as the first one during the fill program and is always controlled as the last one during emptying the reflector borings /U 2.5.4–7/. The KLAK columns within a group are always emptied in such a way that the fill level within the controlled group is about the same.

The reflector rods remain in their normal position in the nominal load operation of the core equilibrium, in which they absorb the reactivity at about 1.2% /U 1/. All KLAK columns are filled up to a fill level of about 1 m to maintain minimum neutron fluence values in the bottom reflector region.

The reactivity is absorbed and released during load alternation operations using the reflector rods and the three KLAK units of the last KLAK group.

The installation depth of the reflector rods is limited for operative reasons; in this case it ensures the availability of adequate shutdown activity if required. This starting limitation is set on site for the then-existing condition of the reactor core (primary core, start-up phase and equilibrium core). The admissible entry depth is determined from the measured values for the reactivity effectiveness and the reflector rods effectiveness as a function of entry depth (S-curve). It is also determined by computing the values for anomalous incidence reactivity and for reactivity recovery as a result of a reflector rod that doesn't fall down and the partial load – zero load – reactivity /U 2.5.5–2/.

The long-term partial load operation at 50% of the nominal load ensures that the reactivity absorption is transferred after about 45 hours to the three KLAK units of the last KLAK group and the reflector rods are withdrawn to the normal position /U 2.5.5–2/.

We have verified, whether the given operating programs for the reflector rods and the KLAK units can

- Maintain an adequate effectiveness of the shutdown elements,
- Limit the reactivity ramps to the admissible value by the reactivity calibration agents (reflector rods and KLAK units),
- Maintain admissible power density values during the normal operation and an initial situation for anomalous incidences.

The limiting of the operative entry depth of the reflector rods maintains sufficient shutdown activity. The procedural method, on adjusting the starting limitation developed on the basis of the then-existing measurements and computations, is acceptable in our opinion. The necessary measurements can be obtained by using a then-existing measuring program for effectiveness of the reflector rods – if necessary by restricting the power operation. The intervals for verifying the starting limitation can be specified only during the process of surplus reactivity in the start-up phase.

The KLAK units have on one hand the operative tasks of forestalling the neutron flow in the lower area of the borings for the KLAK columns and of absorbing the reactivity during start-up and shutdown processes and during load alternation operation. On the other hand the KLAK units display a part of the shutdown system. The maximum fill level of the KLAK columns can be limited to maintain the effectiveness of the KLAK units as shutdown system during each operating condition and during long-term rise in the surplus reactivity.

The reflector rods entry depth limitation during start-up /U 2.5.5-3/ is suitable for preventing the reactivity supply if the reflector rod crashes. In this case there could be a reactivity supply if a reflector rod with greater entry depth and hence greater reactivity absorption between opened KLAK columns falls, where its effectiveness is reduced due to the shielding effect. We accept the mentioned rod starting limitation also as a proof for the shutdown safety of the reflector rods on start-up. We regard the calibration values for rod starting limitation provided by the applicant as relating to the normal position, corresponding to 1.2% absorbed reactivity in the equilibrium core, as temporary, until the effectiveness of provided setting values is proved. It is verified during the constructional planning that the provided setting values for limiting the reflector rods entry depth control the reactivity supply for the presumed fall of a reflector rod.

The operating method of the KLAK unit in groups, where the units of one group are distributed as evenly as possible around the reactor core, is determined optimally considering the maximum reactivity absorption on one hand and minimum rise in the maximum power density on the other. The distribution of the KLAK units in groups maintains a nominal additional axial rise of the power density distribution. We estimate under adverse circumstances (starting with nominal load in the Xenon-free condition) that on starting, the radial azimuthal rise of the power density distribution can safely maintain the temperature limit of fuel element design.

Considering the power density distribution added as a starting condition for the pressure release anomalous incidence, we expect that the temperature limit of the fuel elements can be maintained even during an anomalous incidence on starting. Maximum power density values can exceed those based on the nominal load distribution of equilibrium core on starting, but the afterheat production on starting is lower than in the equilibrium core. Extensive quantitative considerations are not necessary in this case within the design evaluation scope. It is verified for the start-up that the power density distribution and the afterheat production resulting from that is limited in such a way that the temperature limit of the fuel elements is maintained during pressure release anomalous incidence.

2.5.5.10 Summary

The result of our verification of the reactor physical design of the HTR two-module reactor is that the safety regulation requirement of the reactor core can be maintained considering

- Shutdown safety,
- Admissible power density values, and
- The inherent safety.

The reactor physical characteristic quantities serving as boundary conditions for the anomalous incidence analysis are complete and accurate in our opinion.

We have provided instructions for the constructional planning and for the reactor operation as further verification incidences. The main instructions concern the structure and calibration of controlling devices, which control the reactor operation, to maintain the shutdown safety and the power density distribution provided for the anomalous incidences. Further instructions are necessary, because in the supplementary documents the applicant provides details for operating the reactor, which exceed the design limits and it is necessary to partly supplement these.

2.5.6 Thermodynamic Design

Refer to Table 2.5.6–1 for thermodynamic conditions (nominal values) of the primary circuit. The two-dimensional computations by the applicant with the THERMIX program /2.5.6–1/ are the basis for determining these conditions. Essential boundary conditions in this case are the representative core conditions adjusted for the full load operation, for example entry depth of the reflector rods approximately 2.5 m below the upper edge of the core and Xe-equilibrium.

The primary circuit blower supplies the Helium through the annular gap, which is formed from the hot gas pipe and the connecting pressure tank, in the reactor pressure tank. There it is diverted to the bottom area, circulates in the sphere fill pipe and the supporting structure of the bottom plate and then arrives at the 72 cold gas borings of the side reflector. The six reflector rods arranged in the side reflector are cooled using a 1% cold gas by-pass of the total cold gas flow.

The Helium arrives in the cold gas collecting vessel above the core after flowing through the side reflector; from there it flows through the borings of the top reflector, the space above the sphere fill and finally the sphere fill itself.

The Helium enters the borings of the bottom reflector at the end of the sphere fill and arrives at the canals of the so-called mixing device running diagonally direction to the core axis, is mixed radial and flows finally into the hot gas collecting chamber. Form there it is supplied across the hot gas pipe of the evaporator. Subsequently it is fed again in the reactor pressure tank from the primary circuit blower.

Table 2.5.6-1: Thermodynamic Data of The Primary Circuit (Nominal Values)

Thermal reactor power	200 W
Medium power density	3 MW/m^2
Core diameter	3 m
Active core height	9.4 m
Primary circuit pressure	60 bar
Discharge pressure loss	0.68 bar
Bottom reflector pressure loss	0.13 bar
Discharge voids fraction	0.39
Primary coolant throughput	85 kg/second
Bypass through the gaps between the reflector blocks	5%
Bypass for cooling the reflector rods	1%
Medium primary coolant intake temperature (reactor pressure tank unit)	250°C
Side reflector enthalpy rise	10°C
Discharge enthalpy rise	440°C
Medium primary coolant exit temperature (exit discharge)	700°C
Maximum primary coolant temperature (nominal in the core center, intake bottom reflector)	784°C
Minimum primary coolant temperature (nominal in the core border, intake bottom reflector)	660°C
Maximum fuel element central temperature (exit discharge)	830°C
Maximum fuel element surface temperature	812°C
Maximum cladding tube temperature of the reflector rod	600°C
Flow speeds	
– Fill intake	5.5 m/second
– Fill discharge	10.5 m/second