and 5.4.1 (Leaks and Cracks on the Primary Side). The operational details such as, for example, the component loads occurring as a consequence of the heating or cooling, and the illustration in the operational manual can be taken into consideration during the further design and operational development.

We have no reservations with respect to the planned operation modes.

Revision

The HTR module Reactor concept with the continuous fuel element replacement basically allows an uninterrupted operation of the reactor, since no downtimes for fuel element replacements are necessary. However, many of the necessary periodical tests of the operation-relevant safety-relevant plant sections cannot be performed during the full power operation; this is why regular downtimes for revisions are planned.

The kind and range of the repeated tests were described in the sections of Chapter 2 as far as this is necessary as a part of concept assessment with safety relevance. We have described the special aspects of the radiological occupational safety during when carrying out maintenance work and during the repeated tests in Chapter 3.2.3.

The applicant wants to have the neighboring module in full operation, while one module undergoes a revision. Based on this fact, the effects of possible breakdowns in the module working at full operation on the revision personnel should be examined and, on the other hand, it is necessary to find out the admissibility of revision work on common supply arrangements with regard to the occupational safety. The applicant presented a documentation /U 4.2.4/ with this respect.

Events that may damage the health of the revision personnel are, for example, leakage accidents on the primary and secondary side. For the leaks on the primary side are taken measures to minimize the effects on the personnel, such as, for example, the design and arrangement of doors and drainage channels for leaking gases. For secondary side leaks (for example, in the fresh steam pipe) additional limitations of the access to certain plant areas may be necessary.

During the detailed design development, in addition to the conceptual applications mentioned above, there is a need for evidence to be presented concerning the admissibility of the remaining effects on the operational personnel.

As a consequence of the repeated tests during the revision of a module, redundant apparatus of the power supply system are not available for the module for certain time during full operation. During the tests of the 220 V direct current switching station of the reactor protecting system as well as the whole control system, for example, would suffer power failure if the redundant switching station also suffers a breakdown. Because of the limited time for such tests and the very low probability of a switching station breakdown, a simultaneous non-availability of both switching stations is assumed to be highly improbable. The plant is equipped for such cases and the possible effects, which was proven in detail during the breakdown analyses (see Chapter 5).

The prerequisites to be fulfilled during the repeated tests of the type described herein are to be put in concrete terms and in writing during the design development.

On the whole, our opinion is that the HTR 2-module power plant in principle allows a revision of one module during the operation of the second module.

4.2.2 High-speed Shut-off

The high-speed shut-off is started (for a specific module) when the following limit values are reached:

- Neutron flux in the medium range ≥ max
- Period ≥ 20 s
- Thermally corrected neutron flux ≥ 120 %
- Negative sliding limit value of the thermally corrected neutron flux ≥ 20 %/min
- Hot gas temperature ≥ 750 ° C
- Cold gas temperature ≥ 280 ° C
- Throughput ratio of primary circuit to secondary circuit ≥ 1,3 or ≥ 0,75
- Negative sliding limit value primary circuit pressure ≥ 180 mbar/min
- Negative sliding limit value fresh steam pressure ≥ 8 mbar/min
- Wet primary circuit ≥ 800 vpm

In addition, the high-speed shut-off can be initiated manually.

When starting a high-speed shut-off, the following reactor protective measures are taken:

- Retraction of the reflector rods,
- Switching off the primary circuit blower,
- Closing of the stop valves on the side of fresh steam and feed water.

In addition, an operational automatic system closes the blower valve with some delay.

The state of the plant after a manual high-speed shut-off is characterized by the fact that, when the reactor is shut off, the primary and secondary flow is interrupted. At first, the secondary heat is stored in the core and its neighboring structures and, after is then conducted away in slightly increasing quantity over the surface cooler. Insofar as the high-speed shut-off is triggered a defect, greater or smaller deviations from this state can occur. We gave our opinion on this matter in Chapter 5.

Because of the xenon development following a high-speed shut-off and the loss of reactivity associated therewith, a start-up of the plant can be initiated after approximately 1 hour (hot start), if the state of the plant allows it.

The flow processes on the primary and secondary side are determined by the run-out characteristics of the blower and the shut-off times of the fresh steam and feed water stop valves. As a consequence of the natural convection which develops in the reactor core after the high-speed shut-off and the switch-off of the blower, a temperature redistribution and a slowly temperature increase take place in the core; wherein after one hour, a maximum fuel temperature of approximately 900 ° C develops.

For the case wherein the blower valve controlled by the operational automatic system does not close, a natural circulation occurs in the whole primary system which his in the opposite direction to the normal stream direction /U 1, U 4.2-2/. Thereupon, in contrast with the case with the closed blower cover, the helium heated up in the core is transported via the cover and side reflectors to metallic structures, wherein the development of gas temperatures depends essentially from the primary throughput. This throughput amounts in the beginning to approximately 4 kg/s and, after 1 hour still to approximately 1,2 kg/s, which has a consequence a gas temperature of approximately 360 ° C in the area of the lower RDB. According to the data in the safety report, neither the RDB nor the subsequently flown-through components, such as the steam generator and the primary circuit blower, are subjected to an inadmissible load.

The protective measures initiated after a high-speed shut-off have as an objective

- to carry out a nuclear shut off of the reactor by retracting the reflector rods and interrupting the helium flow
  rate and, in this way, limit the core temperatures,
- to limit the temperature changes in the metallic areas as a consequence of the primary and secondary flow, and
- to limit the fresh steam pressure by a delayed closing of the fresh steam stop valve so that a response of the fresh steam safety valve.

Having examined the data contained in both the safety report and in the supplemental materials /U 4.2-1,U 4.2-2/, we have now arrived at the conclusion that the protection objectives have been fulfilled. The crucial data that were used apply mainly to the throughput and temperature gradients at the selected positions and are, in our opinion, sufficient to enable the review of the concept of the plant. In the course of subsequent development planning, it will be necessary to further specify the pressure gradients at the secondary side, as well as the complete data concerning the temperature gradients in the core and the metallic structures for the event of a non-closed blower valve. Furthermore, it must be explained what measures would be taken in order to reduce the fresh steam quantity at the secondary side and which effects would results on the neighboring module.

4.2.3 – Secondary Heat Discharge

As explained before, the secondary heat discharge occurs both during the functional shut down of a module and during the shut down of a RESA in the standard case via the primary heat transfer system and the start-up and shut-down system /U 4.2-3/.

In the event wherein no primary heat transfer system is available, the secondary heat discharge is provided via a surface cooler. Taking into consideration that lost heat is discharged via a surface cooler even in standard operation, no more measures will have to be adopted when putting the plant it into operation. In situations where a cold start of the primary system is required and wherein the secondary heat discharge via the primary heat transfer system is impossible, a cooling may also be effected via the fault cooler of the redundant gas purification plant /U 4.2-3/.

In the event of a long-term secondary heat discharge via a surface cooler after a RESA, the natural convection in core region occurs due to high helium density in the reactor under pressure as a result of different lifting forces. Due to this, as well as due to the heat convection and the heat radiation, heat is transported from the core to the neighboring structures. By interrupting the forced convection and creating a natural convection, a heat distribution from below upwards occurs especially in the core. The heat transportation from the structures adjacent to the core up to the surface cooler then occurs via the heat lines in the RDB extensions, and a natural convection and heat radiation in the helium and air-filled chambers take place.

In the already-mentioned case, the temperature calculations have been conducted by the applicant /U 5.4-6, U 5.4-7, U 5.4-2/ by means of the computing program THERMIX /U 5.4-6, L 104/. Accordingly, maximum core temperatures of approximately 1130 °C are reached after about 6 hours, while using the initial state of 105 % rated power and an allowance of 8 % at post decay power. /U 5.4-6/. Taking into account error influences at the relevant input variables, there is an allowance of 120 K; consequently, the maximum core temperature amounts to approximately 1250 °C. The maximum RDB temperature is achieved with approximately 320 °C after about 75 hours and the maximum core container temperature of about 400 °C is reached after approximately 60 hours.

In the above mentioned event of a non-closed blower valve, the natural circulation in the primary system drops continuously in further course and amounts, after 25 hours, to only approximately 0,15 kg/s. The gas temperature of the helium escaping from the lower side reflector reaches the maximum value of approximately 390 °C after about 8 hours. According to the data of the safety report, both the pressure vessel unit and the other components are insufficiently heated due to low performance. The development of the core temperatures takes a more favorable course as in the case of a closed blower valve, because as a result of the natural circulation in the overall primary system, the heat discharge from the core is further increased due to this additional effect. In the case which was also reviewed /U 1, U 4.2-2/, namely wherein the steam generator continues to be supplied when the blower is switched off and the blower valve is not closed, the primary system circulation is brought to a standstill after approximately 2-3 hours. In this case, comparable core and component temperatures occur as in the case of closed blower valve.

The secondary heat discharge via the primary heat transfer system is contemplated both for a standard operational shut-down as well as for after a RESA. In the latter case, the temperature redistribution, built up after the RESA, is reduced again as a result of the renewed forced convection. Depending upon the time wherein the primary heat transfer system and the blower are put into operation, as well as according to the level of the primary throughput, the gas temperature at the core drops continuously, or else rises first until the original temperature profile is created, and then falls again /U 4.2-3/. In this case, temperatures above the operational hot gas temperature of 700 °C may be attained. Due to the fact, however, that in this case the standard flow direction returns and only the post-decay power needs to be discharged, the temperature load of the pressure vessel unit, as well as other components is not seen as non-critical in comparison with the other cases. In this context, however, we consider that the most advisable is a careful operation; detailed procedures must be discussed within the framework of the operation planning.

In the event of a long-term secondary heat discharge via a surface cooler, both the core and the RDB extensions are heated. The computing program THERh1IX was used in this case to compute the temperature development. We consider this program appropriate for conducting the analysis and otherwise wish to refer to Chapter 2.5.6, in which this program is described and evaluated. Also considered as vital for the temperature calculations is the determination of substance values, which was conducted in a similar manner as in the calculations of a pressure removal failure /U 5.4-6/. We have no reservations with respect to the determination of substance values and in this context wish to refer to Chapter 5.4.1. When determining the maximum core temperature, the effects of the systematic and statistic errors were taken into consideration by the manufacturer. The determination of these effects is appropriate for the problem and complies with the requirements of the KTA Regulations 3102.5 /L 48/.

In the case of the secondary heat abstraction via a surface cooler with a non-closed blower valve, we expect no less favorable or at most a slightly less favorable temperature development as is the case with the closed blower valve.

No comparative calculations for the event of a secondary heat discharge via a surface cooler in a reactor under pressure have been conducted so far. We believe that such calculations are not necessary for the concept review, as both of the maximum temperatures in the core and the RDB have been sufficiently covered by the maximum temperatures of the pressure relief failure described in Chapter 5.4.1

To sum up, we believe that there are no objections as to the planned concept of the secondary heat discharge.

4.3 Start-up Operation

The start-up operation involves the initial operation of the individual components and systems, as well as the entire plant, in compliance with the accepted terms, including the execution of tests and experiments. According to the safety report, the start-up operation should be conducted in four subsequent stages:

Phase A: Preliminary operation tests

Phase B: Initial core charge, initial criticality, and tests at zero power

Phase C: Heat testing

Phase D: Tests of the capacity range, contractual trial run

With a view to the technical prerequisites and the safety terms, it is expected that before a high level part of the facility or system is put into operation, the required partial or supply systems will be put into operation in order to facilitate a safe continuation of the starting process.

This also applies to the approval of the individual starting operation stages, as well as to the appropriate Stage D capacity ranges.

The start-up operation starts after the assembly of a component or a complete system, including the preliminary operation tests of Stage A. They comprise the complete start-up operation of the plant's individual parts and systems. The preparatory tests start with a check of the system and is over after the process technical start-up operation is completed, during which the safe and correct operation of the individual components, as well as the correct overall functioning of the systems according to planning and design are documented.

The filling of the primary core elements is conducted under air atmosphere, with an opened reactor pressure tank. In order for the core filling to be always conducted within the efficiency range of the reflector rods, the filling is first produced by moderator elements, up to approximately 1 m below the core center. In order to reduce the fall speed of the core elements to acceptable levels, the HE feed tube is fitted with extension pieces with built-in mechanical brake sections. The core elements are filled onto the moderator filling in the primary core mixing ratio, with the smallest critical compound being built in adequate partial steps. The filling process is controlled by measuring the sub-critical multiplications, and the filling quantity per one step is determined from the preceding step count increase

As soon as the „smallest critical mass" is formed, the reactivity equivalent of the reflector rods and the KLAKE units is determined at zero throughput according to the inverse kinetic method. In the course of this first step of the core loading is provided an additional load instrumentation, which consists of counters carried along the core filling surface. In this way, it should be possible to attain minimal flow values for sub-criticality control.

The second loading of core element mixture depends on the comparison of the previously computed with the measured „smallest critical compound." If they correspond, the loading with the predetermined mixture ratio is continued. If the deviation is too great, the mixture ratio is corrected in order to achieve the required cold redundant reactivity with filled core at any case.

Any subsequent loading is again conducted in partial steps, with the reflector rods partly extracted, wherein the required moderator elements are extracted from the lower core region, so that the reduction and filling of the zone containing the fuel can be conducted simultaneously. After each partial step, the following zero-throughput tests should be conducted:

- The detection of any excess activity of the core,
- The determination of the switch-off values of the reflector rods and KLAKE units,
- The determination of the sub-criticality with the non-filled most reactive KLAK pillar and completely extracted reflector rods

If the sub-criticality measurements show that full provided excess activity must not be attained, the core
element mixing ratio of the remaining loads should be adjusted so that a sufficient sub-criticality is available for the full initial core.

The planned heat test is conducted first under nitrogen and subsequently under helium. During the test under nitrogen, the primary circuit is first heated via the energy loss of the primary circuit blower to cool gas temperature of approximately 250 °C. During this test, the functioning of systems crucial for the primary side operation, as well as the primary circuit instrumentation are tested at different pressures and temperatures. In this state are also determined the excess activity of the reactor core at various medium core temperatures, as well as the efficiency of the shut-off elements according to the nitrogen compensation method. This is effected by adding nitrogen to the primary circuit, up to the maximum reactivity linkage of 2 R, in order to limit the failure potential during an involuntary pressure drop.

A complete and fast degassing of nitrogen, penetrating into the graphite pores, such as without the use of reflector rods, would result in a temperature increase of 40 to 80 K /U 4.3-1/.

When conducting tests under helium, the primary circuit is restored to the cold gas temperature with the help of the blower and initially run at an operation pressure of approximately 60 bar. In this way, it is possible to check the coordination of components and systems, as well as of the helium purification plant, under running conditions. The tests, conducted under nitrogen, are partly repeated. Should it become evident in this stage of the start-up operation that the planned rated power cannot be attained, the reactor is shut off and, insofar as the cold sib-criticality of the reactor allows it, the core element filling is turned over. Thus, the core element composition is changed according to the additional demand for redundant reactivity. If such step is made impossible or only partially possible because of the current sub-criticality, the reactor is first run at reduced capacity and reduced gas throughput temperature. Due to the ensuing build-up of additional fission product, the reactivity demand for attaining a rated load is reduced, thus enabling a step by step increase of the capacity in the running-in stage.

The tests, measurements and experiments in throughput range should be conducted in three different throughput levels, from 0 % to 30 2, 30 % to 80 %, and 80 % to 100%.

According to the security report , the following important tests and measurements in one or several throughput ranges are planned:

- Determining the reactor power and calibrating the neutron flow signals,

- Checking the progress performance, especially that on the primary side control and regulating devices,

- Testing the screening devices in the accessible rooms,

- Chemical and radio-chemical inspection of the primary cooling agent.

Furthermore, the performance of the entire plant must be checked under running conditions and increased power. This may be the case both under stationary operating conditions, as well as under the targeted introduction of faults, in order to be able to assess the performance of the entire plant in conformity with the design in the standard operation. In this connection, the following experiments are planned:

- Sudden and ramp-shaped changes of load, start-up, and shut-off,
- Reactor fast shut-off,
- turbine fast shut-off,
- Failure of own consumption supply and changeover to emergency power supply
- Failure of primary circuit blower,
- Failure of feed water pump,
- Failure of the process steam system.

Following the completion of the power tests, the test operation of the HTR 2-module power plant will follow. This test will be completed upon delivery to the operator.

The start-up operation activities are conducted according to the start-up operation programs, which are system-oriented in stages A to C and stage-oriented in stage D. The IHS programs are amended by start-up operation instructions, which specify comprehensive IHS activities and frequently recurring activities. Those of the stages which are crucial for checking the entire plant are further detailed in start-up operation schedules, which contain both the sequence of the start-up operation activities, as well as their dependence and approximate time slope. Apart from that, a schedule has been designed, which specifies the objectives that should be achieved when starting up the individual systems. The start-up documents comprise document sheets, acceptance records, result reports, and results records.

The start-up operation of the HTR module power plant described in the security report has been based on a procedure previously used with success for start-ups as well as that of the fuel elements during planned failures By extending the fuel element feed tube, in which are built-in the mechanical breaking sections, the fall velocity of the core elements can be slowed down and thus the acceptable core elements impact velocity can be fulfilled. The effect and functional security of the breaking sections can be documented in the development design.

The presentation of the start-up operation of the HTR module power plant shown in the security report is oriented toward the procedure which has already been successful during start-up of other plants /L 89, L 90/. A sequence of the first criticality of the reactor and the heat sample operation, which deviates therefrom, may be caused by the module-specific core construction.

We processed, in particular for the time span "start of loading – reaching of nominal load," the following testing points:

- maintaining of the allowed mechanical load of the core elements,
- monitoring of the under-criticality during loading,
- monitoring of the under-criticality to determine switching safety when using the KLAK system,

- maintaining the allowed loads in the reactor pressure vessel built-in components and the fuel elements during expected disruptions.

The monitoring of the sub-criticality during loading can be ensured by the additional loading instrumentation, by the neutron source, as well as by external core instrumentation. The efficiency of this check system, or of the minimum required counting rate, as well as the correlation between the counting rate and the multiplication factor of the reactor core can be documented in the development design. By separating the core element filling into filling quantities, which are determined according to the reactivity increase over the preceding filling step, it is possible to ensure that the distance from the critical stage can be sufficiently reduced.

The monitoring of the reactor core sub-criticality /U 2.13-5/, following each partial filling step is not sufficient to check the shut-off security when employing the KLAK system. Not only is it possible to expect the failure of the reactivity most efficient KLAK pillar, but the change of reactivity during the changeover from the „core element filling under loading" to the „reactor core under operating conditions," as well as the reactivity supply are contingent on the faults and must also be taken into consideration.
As for the reactivity change, the following must be considered:

- the reactivity supply due to out-gassing of nitrogen and water from the core element filling and the reflector
graphite, due to the combustion of the graphite contaminants during the initial part of the running-in stage,
as well as due to the changeover from air or nitrogen filling to helium filling of the reactor core,

- the reactivity loss due to the reduction of the filling factor from loading to circulation operation.

The reactivity increase, observed by the applicant, due to the fast degassing of nitrogen during the changeover to helium atmosphere in the reactor core /U 4.3-1/ cannot be regarded as the only failure during the loading and the reactor physical measurements. Considering that the reactor physical parameters, such as the dependency of the reactivity on filling factor, as well as the efficiency of the shut-off values during loading, change considerably, it must be proven that the effects described in Chapter 5.2 sufficiently cover the expected reactivity failures during loading.

Any extra measures to be adopted must be specified. We believe that the proof is admissible, while it is necessary to adopt such extra measures such as blocking free fall paths of the reflector rods, or the integration of the loading instrumentation into the input signals of the reactor system.

Not mentioned in the specification is the fact that the planned rod drive restriction and the reactor power restriction during the putting into operation stage must also be checked. We, however, take this for granted. It is also not quite clear how the plant tests should be conducted with regard to half-load plants interconnected at the secondary side, namely if the tests are reduced only to such a plant, with or without the second plant in operation, or if they should be conducted in both plants in parallel. Taking into consideration the requirement of the BMI security criteria /L 6/ for a possibly fault-free operation of the plant, we believe that, considering the planned concept of the plant, the tests of the mutually connected components at the secondary side, including the high level block control, as well as the subordinate secondary side controls should be carried out. In this connection, the operation condition with only one reactor in operation, yet with several consumers at both secondary sides, should be considered.

We believe that the above-discussed viewpoints are concept-relevant, due to the fact that only the final assessment of the integrity, as well as of the details of the start-up operation tests are relevant for planning the operation. To sum up, we consider that the planned procedure for starting up the HTR module power plant is practical and in compliance with the state of the art. As for the concept of the process of starting up the plant in the described manner, we have no objections.

5 Failure Analysis

5.1 Introduction

Integrity of the Failure Analysis

Pursuant to Section 7, paragraph 2 of the Atomic Act, core power plants must adopt adequate measures, in accordance with the current state of the art, to prevent damage due to construction and operation of the plant. It is absolutely crucial that in case of activity emission during a failure, the planned parameters pursuant to § 28 paragraph 3 of Radiation Protection Regulation L 2/ must not be exceeded. In order to balance the security concept, it is further necessary to see that in case of such failures, the radiation to which the staff and the surrounding area are exposed is not only reduced, but also that the following fundamental protection objectives are fulfilled:

- the integrity of the barriers against activity emission must be observed up to the required extent,.
- the functionality of the systems of technical importance must remain intact within the required up to the required extent

In order to prove the observance of the above requirements, all failures must be checked which are relevant with regard to the above security objectives.

In Tables 2.5-1 and 2.5-2 of the security report, the applicant has specified all failures, from which the HTR 2-modul power plants are protected due to their design. Apart from that, Table 2.5-3 also specifies such events, against which risk-reducing precautionary measures have been adopted.

This list has been compiled by the applicant by meaningfully applying the principles of failure guidelines for core power plants with pressurized water reactors /L 6/, as well as by transferring these to the concept of the plant. The individual failure events have been specified in the tables according to the high level and plant- independent classification indicated in these guidelines

In an additional document /5.2-1/, the applicant has explained its standpoint concerning events outside the design framework in order to illustrate the existing security reserves of the HTR module power plant, as well as the specific, inherent security features of such a plant.

In order to obtain a complete list of the observed failure triggering events, the documents /L 14, L 18, L 68, L 72/ in Table 5.1-1, Part I, have been compiled, which contain all the events to be considered within the course of the approval process for light water reactors. This list has been arranged according to the BMI for a LWR standard security report /L 18/. Several triggering events observed in light of water reactors have been eliminated in the HTR module due to its completely different design. These have been indicated in the right column. In the remaining cases, specified in the corresponding column, reference is made to the respective section of the expertise, in which the failure is described.

Aside from this, the discussions concerning the approval process for the HTR currently underway in the Federal Republic of Germany have also been considered /L 121, L 80 to L 83/.

Special attention has been paid to the water gas generation in the event of a fracture in the steam generator hot pipes. In AVR, such event had to be examined in detail, due to the fact that AVR is fitted with a steam generator, which is integrated into the core vessel /L 85/. THTR faces similar problems. In the event of a hot pipe fracture, the respective steam generator must be isolated in such a case, the remaining steam generators take charge of the heat discharge. Part II of Table 5.1-1 contains a list of several other „events to be considered" in HTR facilities. Also considered have been all the triggering events mentioned in risk considerations /L 128 to L 131, L 84/.

All in all, it may well be stated that the list of design failures presented by the applicant in Chapter 2.5 of the security report is complete and that the demarcation of the hypothetical range has been correctly selected. A sufficient risk reducing measures have been adopted by protecting the plant from external civilization-dependent effects, as well as by providing the option of the external surface cooling agent supply and power supply of the emergency control room, while taking into account the special design characteristics of the HTR 2-module power plant.

No standpoint is taken within the framework of this assessment with respect to the hypothetical development of failures specified in document /U 5.2-1/

Procedure for Failure Analyses

In the below sections, the failure development resulting from failure triggering events has been described. In this context, deterministic requirements based on following high level rules and regulations have been considered:

- the core power plants security criteria /L 6/,
- the design of the security criteria for high temperature reactors /L 7/,
- the principles governing the application of single failure concept /L 8/,
- the Failure Guidelines /L 11/,
- the RSK Guidelines /L 10/,
- the KTA Regulations 3501 /L 49/.

From this follow the below boundary conditions for the failures to be examined:

- the failure of the 1st criterion for reactor protection impulses
- the adverse operation condition as initial condition,
- the single error or, if applicable, the maintenance event for the systems required to contain the failure,
- the non-consideration of operation systems required for the immediate failure containment.

In order to judge the observance of the above crucial protection objectives, it is necessary to check which distance must have the components, systems, and buildings from their design limits. In the following sections, special protection objectives have been specified with their respective boundary values. These boundary values are to be used as evaluation criteria.

Computer Programs

The computer programs used for the failure analysis of the HTR modules have been verified. In most cases, this verification is conducted at AVR or in suitably equipped testing facilities. The tests conducted to verify the computer programs are known to us. Reviewed from the viewpoint of security have been, for example, the experimental cooling agent loss failure test, as well as the dust emission test during a pressure release failure /L 86, L 87/.

In the following chapters, we have reviewed the individual computer programs, used by the applicant, as well as their experimental security and verification.

Where necessary, our own computations have been conducted, for example, to establish the maximum steam generator leakage when there is a fracture in the steam generator hot pipe, the reactor physical design calculations, the temperature calculations to establish the maximum fuel element and the component temperatures, especially in case of pressure relief failure, as well as the pressure calculations to establish the building structures load.

Table 5.1-1: List of failure triggering events to be monitored in light water reactors and high temperature reactors.

I. Events to be Monitored in LWR

Number of Evaluation Chapter Dealing with Said Event	
Release of the most effective control element, or the most effective control element group or bank from different plant conditions	5.2.1
Ejection, or falling out of a control element with taking into consideration adverse starting conditions for power, power distribution and reactivity supply	5.2.2
Defective falling or defective insertion of a control element	5.2.2
Connection of a cooling water circulation pump	5.2.3
Cold water supply in the reactor cooling system from adjoining systems	5.2.4
Pressure changes in the reactor cooling system	5.5.8
Involuntary reduction of the boron content in reactor core region Constructively not applicable because there is no boron in the primary circuit	
Power control defects (control failure), - resulting from adverse operating condition; adverse assumptions regarding control failure (SWft)	5.2.3
Insertion and start-up of a fuel element - in incorrect position of fuel element loading)	5.5.2 (corresponds to disorders in the fuel element load)
Turbine rapid closure with by-pass station opening	5.3.2
Turbine rapid closure without by-pass station opening - (for example, condenser vacuum loss)	5.3.2
Main heat drain failure due involuntary closure of the fresh steam insulation valves	5.3.3

HTR Module Table 5.1-1: continued

I. Events to be Monitored in LWR

Number of expertise chapter dealing with such event

Number of Evaluation Chapter Dealing with Said Event

Failure of the coolant circulation pumps 5.3.1

Main coolant circulation pump blocking or break 5.3.1

Own supply failure (emergency power supply failure) 5.3.5

Emergency power supply failure with operation leakage from primary circuit not applicable, primary pressure below secondary pressure

Not applicable, primary pressure lies under the secondary pressure
Failure of the main feed pumps, or incorrect closure of feed water valves 5.3.4

Break in the main feed water line 5.4.2.1

Leak in the feed water line, or in the steam generator sludge settling pipe in circular ring room not applicable due to circular ring room absence

Fresh steam drainage failures:

- Malfunctioning of the feed water or fresh steam system with regard to increased or reduced heat drainage, or pressure control failure 5.3.3

- Involuntary opening of valves (for example, by-pass valves, relief valves, safety valves)

Steam generator leakage 5.4.3

Heating line break (with or without emergency power supply) 5.4.3

Leakage and breaks in the fresh steam line system 5.4.2.2

- Small leaks, cracks and breaks of the connection lines

- Defective release of a fresh steam safety valve with steam generator heat tube damage

- Break in the fresh steam pipe behind the external closure valve with damage of the steam generator heat tube

Table 5.1-1: continued

I. Events to be Monitored in LWR	Number of Evaluation Chapter Dealing with Said Event
- Break of a fresh steam pipe before external closure valve, or in circular ring room	
- Break of a fresh steam pipe impossible to close (undercooling transient)	
Leakage and breaks in primary circuit:	
- Break of a reactor cooling agent line in safety vessel	5.4.1
- Small leaks from pressure conducting casing and break in connection lines. Defective behavior of valves	
- reactor cooling line break outside safety vessel	
Operating transients with subordinated rapid shut-off system failure (ATWS)	5.5.1
Failures of fuel elements handling and storage	5.5.2
Collapse of heavy loads	5.5.3
Water loss in fuel element storage containers storage	not applicable, dry
Failures and leaks in ventilation and combustion gas System	5.5.4 combustion gas system not available
Failures and leaks in sewer system or in contaminated water container	5.8.1.6
Generation of projectiles (for example, rotating parts explosion, turbine failure, valve stem acceleration, container burst)	5.5.6
Fire and explosions inside the facility	5.5.7
Failures of plant control system	5.5.8
Flooding inside the facility	5.5.9

Table 5.1-1: continued

I. Events to be Monitored in LWR	Number of Expertise Chapter Dealing with Said Event
External impacts	5.6
- Earthquakes	
- Aircraft crash	
- Explosions	
- Fire	
- Floods	
- Other external impacts (for example, chemical pollutants, whirlwinds, thunder-stroke, damage due to undermining, land subsistence)	

II. Additional Events to be Monitored in HTR Plants	
Primary circuit flooding	5.2.5 and 5.4.3
Massive air invasion into primary circuit	not applicable (see Chapter. 2.6)
Condensation in fuel element filling	5.2.6
Helium plant facility failure	5.5.5
Leakage between hot and cool cooling agent pipe	5.4.1
Surface cooler failure	5.3.5
Ceiling reflector crash	not applicable (see Chapter. 2.5)
Pellet exhaust tube break	not applicable (see Chapter. 2.6)
Nitrogen leak from the core during putting into operation tests	4•3
Interaction of Multi-module plants	5.7

5.2 Reactivity Failures

Selection of the Monitored Events

Chapter 5.1 of this expert evaluation contains all possible triggering events that may eventually result in reactivity failures. Dealt with below is the progress of the events (reactivity failures). In Chapter 5.5.1 is described the failure of the rapid shutdown system.

The assessment of the progress of the events has been based on calculations specified in the security report and the annexed documents. According to the KTA-Regulation 3501 /L 49/, each failure is to be traced back to an adverse operating condition of the plant due to the effect of such a failure. When analyzing the progress of events for the HTR module, the standard operation of the plant is considered first. In additional parameter studies, the manufacturer examines the effect of adverse operating conditions. These are dealt with in individual failure-specific sections.

When analyzing the reactivity failures, no additional failure of the power supply for own consumption is to be presumed. This failure results in the failure of the blower and the dropping off of the switch-off rods. As a result, the reactor is switched-off. However, individual active system errors (such as KLAK-pillars defective drive), which aggravate the failure are expected and analyzed. Analyses of the balance core were carried out by the applicant.

Contained in the documents /U 2.13-5, U 2.5.5-20/ is the description of the process of the initial core loading, as well as the check of its reactivity. Failure analyses of the condition of the balance core reactor after several years of operation have been conducted, starting from the initial core and reaching the running-in stage. Due to the combustion, the basic reactor physical properties of the balance core after several months operation are to such extent approximate that the impacts on the initial core failure behavior, described below, only apply to the first part of the run-in stage.

- Release of Control Elements and KLAK Pillar

Considering that initial core redundant activity is lower and the load switchover range is limited to the 90-100 & range, control elements in rated load condition are within the lower efficiency range, resulting in lower reactivity ramps than in the balance core. Apart from that, the reactivity reverse effect is more intense due to the determining initial core reactivity factors, as well as due to the increased proportion of retarded neutrons; as a result, failure effects in the balance core may be considered representative.

- Run-up of the Blower

In case of blower triggering failure, the reactivity supply is effected faster than in the balance core, due to the fact that the determining reactivity factor is pushed further into the negative range. Contrary to this, the rapid shut-off is effected at an earlier moment; as a result, the power progress integral, which determines the temperatures in the reactor core and the RDB extensions, does not significantly increase, and the design temperatures of the fuel elements and the RDB extensions are not achieved. The same applies to the cold gas temperature reduction.

- Densification of the Fuel Element Filling During an Earthquake

The reactivity supply at increased filling factor in the initial core is higher due the increased neutron leakage, determined, inter alia, by the filling factor of the fuel element filling. Instead, he neutron kinetic parameter proportion of the delayed neutrons and the negative feedback of the determining reactivity factor in the initial core has a more absorbing effect. In this way, a significant increase of the reactor power with respect to the analysis results to the balance core – also without taking into consideration the 1st rapid shut-off criterion – need not be provided.

- Water Penetration into the Primary Circuit

Due to the substantially higher moderation grade in the initial core (50 % moderator elements in the pellet filling), there is no significant reactivity increase when water penetrates into the primary circuit. This has been proved by our own calculations. We thus consider the balance core failure analyses as sufficient.

Considering the above reasons, it is quite sufficient to review the prevention of damage of the plant with respect to the balance core analyses, when analyzing the concept.

Assessment Criteria

Pursuant to the rules and guidelines specified in Chapter 5.1 become evident the following requirements:

- The reactivity stroke and ramp of reactivity controlling facilities should be restricted, in order to comply with the limit values specified in Section 6 in case of erroneous operating instructions /L 6/ criterion 5.3, /L7/ criterion 3.4.

- The reactor core must be designed so that, the power excursions due to the feedback characteristics are prevented so that any damage significant from the viewpoint of security so not occur /L 6/, /L 7/ criterion 3.2.

- For each event which should be contained by the reactor safety system, at least two impulse criteria should be available /L 6/, /L 7/ criterion 6.1, /L 49/.

- The reflector rods must contain any failure induced reactivity feed even in case of a reflector rod failure /L 6/ criterion 5.3, /L 7/ criterion 3.4.

With running primary circuit blower, the below temperatures used for designing the components must not be stationary exceeded:

- The maximum medium heating gas temperature (radially averaged) at forced circulation 760 °C. Taking into account the cooling between the core throughput and the steam generator input, the design temperature of the superheater tubes of 750 °C /U 2-6.4-/ is not exceeded.

- The maximum average cold gas temperature 350 °C. In this way, the design temperature of the pressure vessel unit of 350 °C is not exceeded.

A short-term failure-conditioned lesser excess of the design temperature is acceptable. This aspect is dealt with in the assessment of the pressure vessel unit.

With switched-off blower (after a AESA), the design of the reactor core and that of the systems must guarantee that in all the reviewed reactivity failures, a maximum fuel element temperature of 1200 °C is maintained.

To calculate the radiation doses, a constant particle fraction is used, which his valid up to 1200 °C. A low excess (up to 50 K) of this temperature for several fuel elements does not result in a change of the calculation activity inventory.

Generally, the active cooling of the reactor is avoided after a failure conditioned RESA. In the case of the operating method specified in the security report, this is the case during the first hour after a RESA. Heat from the primary cell is then abstracted via the surface cooler, similarly as during the reactor power operation. Afterwards, the reactor is cooled via the main heat transfer system as well as by starting up and shutting off the system. Further possibilities for cooling exist via the second module water steam cycle, as well as via the failure water blowroom condenser. The maximum failure temperature is thus arrived at one hour after the RESA.

In case no possibilities of heat discharge from the reactor are available (for example, due to missing own consumption power supply), the active cooling of the reactor may be abandoned by continuously discharging heat from the primary cell via the surface cooler. In this case, the fuel element temperatures, as well as those of the pressure vessel extensions reach their maximum values only after several hours after a RESA, and then continue to fall.

Analyses Methods

The applicant has used its computing programs RZKIND and ZKIND to analyze reactivity failures. These programs are described in the report /U 2.5.5-5/. They differ essentially only in the geometry, not however in the computing methods. Thus, only the two-dimensional program RZKIND is considered, which shows the reactor core in two-dimensional cylinder symmetric geometry.

Essentially, the program consists of three parts, which are called up one after the other in each computing step:

- Neutron physical part for computing the power distribution,

- Thermodynamic part for computing the solid substance and helium temperatures,

- Neutron physical part for computing the feedback.

In the first neutron physical part, the neutron flow in the fuel element filling is computed according to the diffusion equation. The required single group effective cross sections are updated in each computing step of the second neutron physical part. By means of the power distribution, the thermodynamic computing part determines the temperatures in the fuel element filling in the ceiling, floor, and side reflector.

The cooling gas throughput and the cool gas temperature are input variables; the hot gas temperature is calculated. Comparison calculations between the programs THERMIX and AZKIC7D confirm the correctness of the thermodynamic part /U 2.5.5-7/.

The reactivity reactions are considered in the third part by altering the effective cross sections. The polynomes are used to demonstrate the feedback and neutron leakage established by means of extensive calculations in the program system ZIRKUS. Also considered is the effect of the xenon concentration, as well as the position of the switch-off elements.

Comparison calculations are used to secure the development of the computing program RZKZND from the program system ZIRKUS. /U 2.5.5-7/ and /L 137/. The calculation of the control characteristic and the KLAK pillars by means of RZKIND is verified by comparison calculations with the three-dimensional program MOCA /U 2.5.5-7/.

We believe that the distribution of the pressure vessel extensions and the pellet filling into part volumes, described by the applicant in the Report /U 5.2-4/, is sufficient.

Apart from the described comparison calculations with another of the applicant's computing programs, the verification of the computing programs RZKIND and ZKIND was conducted by means of resulting significance-relevant calculation. At AVR, the reactivity dynamics change (for example, the changed reflector rod efficiency) was examined during the changeover from high enriched to low enriched fuel elements. The reactivity development during this changeover was monitored under operating conditions

When examining the reactor dynamics, the reactivity transients at AVA were run by shifting the reflector rods or changing the blower speed. These experiments are described in the publication /L 135/. These transients were checked by the applicant by means of comparable computing programs /U 2.5.5-7/ and /L 136/. The conformity between the two measured transients and the calculation values is good

A comparison of the experimentally determined and calculated temperature coefficients shows that these coefficients can be determined sufficiently accurately by means of the AZKIND program. /U 2.5.5-7/ and L 137/.

We believe that with the RZKIND program, all the effects on the progress of the reactivity failures are considered. The computing methods are in accordance with the state of the art.

To sum up, it may well be noted that there are no objections with respect to the computing methods used by the Applicant to analyze the reactivity failures. In the following chapters, these computing methods are no more considered; the applicant's computing results have been considered as the basis for this evaluation.

5.2.1 Defective Extraction of Reflector Rods or Small Pellet Switch-off Elements

5.2.1.1 Full Load

The defective extraction of reactor rods or small pellet switch-off elements *KLAK pillars) results in an involuntary reactivity feed. The reactivity feed rate is determined by the initial insertion depth, as well as the extraction speed of the reflector rods, or by the filling height and conveyance time of a KLAK-pillar. Ruled out due to the design is the simultaneous backfeed of several KLAK pillars by limiting the conveyance flow.

In the case of reactor stationary full load operation (xenon balance), the redundant reactivity of 1.2 % is compensated as reactivity advance for load change by the reflector rods. After restarting the reactor from light load, larger redundant reactivities are temporarily bound by the reflector rods (for several hours before xenon balance is achieved) The maximum redundant reactivity to be compensated in this case amounts to 2.5%. The insertion depth of the reflector rods is constrained by means of a limiting device to prevent an excessively deep position. /U 5.2-3/. By suitably adjusting the limit value, this device can prevent the exceeding of the maximum redundant reactivity to be compensated. The determination of the limit value is described later within the framework of the operation analyses.

In the case of a stationary load operation (xenon balance), the KLAK pillars have a calculated minimum filling, which reaches approximately one meter below the core bottom edge and they have practically no influence on the reactivity. The redundant reactivity control in the balance core is conducted mainly via the circulation of partially combusted fuel elements, via the extraction of the fuel elements, which achieve a target combustion, as well as via the addition of fresh fuel elements. During the running-in stage, the absorber and the moderator elements are additionally withdrawn. The load changes should be controlled by means of reflector rods. Taking into consideration that the efficiency of the reflector rods is limited, the redundant reactivity during the start of the reactor from the hot xenon-free condition, or during load changes where the required reactivity rise would involve an excessively deep insertion of the reflector rods (violation of the required switch-off reactivity), must be compensated by the partially filled KLAK pillars. In such a case, the maximum xenon reactivity of 3.2 % is to be compensated by the small pellet switch-off elements.

The progress of the failure "extraction of all reflector rods at maximum speed during full-load balance state (210 MW)" is described in detail in the safety report. In this case, an initial power of 105% is considered. In long term (at hour range), a higher reactor power is prevented by means of a power-limiting device. In the medium term, however, is possible a higher than 105% integral reactor power /U 2.5.5-1/.

At rated blower power and a cold gas temperature of 250 °C, the stationary hot gas temperature (average gas throughput temperature) is 722 °C. In accordance with the partial-load diagram it is thus above the full-load value of 700 °C. Starting from this initial condition, all the reflector rods are continuously extracted from their rated position (app. 2.5 m below the core top edge) at a computed extraction speed of 1 cm/s. The possible reactivity rise at this initial position amounts altogether to 1.2 %; however it is not achieved due to rapid shut-off.

Due to the reactivity feed, the reactor power and the gas temperature are increased. An intervention of the control system to reduce power or temperature is not considered. It is assumed that the cold gas temperature remains constant. This assumption is conservative; it delays the response of the second reactor system excitation

After 12 s, the rapid switch-off criterion of the "thermally corrected neutron flow of more than 120 %" is achieved. With the presumed failure of this signal, the reactor power rises up to a maximum value of 210 %, bur is limited as a result of negative reactivity feedback due to increased temperature. After approximately 80 s, the second rapid switch-off criterion "hot gas temperature of more than 750 °C" responds. The hot gas temperature response time of 10 s is included in these 90 s.

Due to the rapid switch-off response, the reflector rods fall in, the primary circuit blower is switched off, and the steam generator on the side of feed water and steam is shut-off. In the analyses, it is assumed that, with the RESA, the reflector rod does not fall in. The falling in of the remaining reflector rods and the switch-off of the blower effect a more rapid reactor power loss, wherein 30 s after the rapid switch-off (RESA) , the reactor power drops below 10%.

Immediately after the RESA, the mean hot gas temperature, which has been radially averaged at the core throughput, reaches its maximum value of approximately 760 °C; maximum fuel element temperature reaches the maximum value of approximately 950 °C. In this connection is taken into consideration a conservative allowance with regard to bound reactivity and the switch-off elements control characteristics.

In an extensive parameter study /U 5.2-4/, the applicant examined the "extraction of all reflector rods" under different boundary conditions. Examined in this context have been the influence of changes in the throughput values on reactor power and hot gas temperature, as well as the changes in the temperature coefficients and the simultaneous extraction of KLAK-pillars. The parameter variations result in nearly the same results as the above-described calculation by the applicant in the safety report. This is due to the fact that even in case of an increase of the possible reactivity increase, the reactivity actually supplied during a failure is limited by the rapid switch-off response to a value specified in the above analyses.

Through the second rapid switch-off criterion "hot gas temperature of more than 750 °C", the hot gas temperature is limited to a value of about 760 °C. Thus, the initial temperature changes have almost no influence on the maximum gas temperature. As a result, the maximum temperature of the fuel elements is also limited. An adverse initial power distribution may however result in maximum temperatures increase, whereas the power maximum with deeply inserted reflector rods is shifted into the lower hot core region.

The highest fuel element temperature results from the assumption that with a xenon-free core, all reflector rods and a KLAK-pillar are extracted at full load. Due to the fact that in this case, the response of the 2nd rapid switch-off criterion "negative gliding limit value of the thermal neutron flow" is not ensured, the maximum fuel elements temperature at the time of the RESA increases with the switch-off of the then activated criterion "hot gas temperature higher than 750 °C" by approximately 60 K vis-à-vis the reference event, described in the parameter study /U 5.2-4/.

The failure "extraction of all reflector rods at maximum speed and full load balance" has been examined, taking into account the adverse operating conditions of the plant, with respect to

- integral reactor power,
- reactivity coefficients,
- power density distribution,
- hot gas temperature.

Thereat, an integral reactor power of 105% at the beginning of the failure was considered. The designed power-limiting device only becomes effective over long term (within the range of hours). In our opinion, the power increases must be prevented even from the short-term perspective. The applicant must prove that the power-limiting device is able to limit the reactor power any time, also with regard to possible a measuring error, to the maximum of 105 %, or that a short-term excess of 105% of the reactor power is acceptable.

In its analyses, the applicant documented that an increase of the stationary compensated redundant reactivity has almost no effect on the result. Similarly, a model tailored variation of the steady hot gas temperature has practically no effect on maximum temperatures reached during a failure.

An additional individual error is not to be expected due to the fact that, with xenon balance and full load, all KLAK pillars are nearly extracted and only the reactor protection system, designed to protect against individual errors, is thus required. Prior to reaching the xenon balance, the KLAK pillars may be partially inserted. The involuntary extraction of a KLAK pillar was imputed as an individual error.

The supplied reactivity is limited by the reactor protection system with two physically different available rapid switch-off pulses. In the case of the reactor rapid switch via the second response criteria wherein the failure of a reflector rod is imputed, the following values are arrived at the moment of the rapid shut-off in unfavorable case:

- maximum fuel element temperature of 1010°C at RESA for the adverse failure progress at full power,

- maximum medium hot gas temperature of 760 °C,

- maximum medium cold gas temperature of 250 °C (maintained constant in the calculation).

The above named limits for the design in case of forced circulation are thus maintained.

After the RESA, heat is discharged via the surface cooler in the first hour. At this moment, the maximum fuel element temperature increases by 40 K. In the event of the most adverse course of failure, this results in the highest value of the maximum fuel temperature of 1050 °C, when the secondary heat discharge is assumed by the main heat transportation system after approximately one hour.

In the event of an assumed failure of the main heat collector and a secondary heat discharge via the surface cooler, the temperatures first continue to rise. In addition to the calculation in the security report, the applicant has calculated the long-term temperature courses after a reactivity failure /U 5.2-4/. In this case, the extraction of all reflector rods is examined under full load with a power integral of 180 full-load seconds up to the RESA moment. From this calculation it follows that, after such a reactivity failure, the maximum of the maximum fuel elements temperature is higher by 100 K than after a RESA in the rated operation.

In the case of a RESA from standard operation and the reactor under pressure, the maximum fuel element temperature increases from approximately 88O °C up to 1130 °C; the maximum gas temperature reaches nearly the same value. In the event of an adverse course of failure under full load, the power integral amounts to 200 full-load seconds up to the RESA. When applying the calculations in the security report and /U 5.2-4/, it may well be presumed that, in an adverse case, a maximum fuel element temperature reaches 1240 °C at the upper core edge. Only several fuel elements of the pellet filling reach this maximum value. This temperature is arrived at several hours after the RESA. It runs that much below the limit value of 1620 °C that the effect of the measuring errors or calculation inaccuracies does not need to be examined. When determining the maximum fuel element temperatures, the calculation inaccuracies lie under 100 K /U 5.2-4/.

The failure "extraction of all reflector rods at maximum speed and full load balance (210 MW)," described in the safety report can be considered a covered design failure.

The temperatures of the pressure vessel extensions and the pressure valve itself are practically unaffected by the short-term temperature increase between the start of a failure up to the RESA; they maintain their standard operation stationary value. In the case of the observed reactivity failure, practically the same courses of temperature apply to these components as to a RESA during standard operation. Thus, the limit values for the design are not exceeded.

In our opinion, it is possible to prove on the basis of the reactivity failure observed here that the secondary heat discharge can also then be ensured without exceeding the design limit values if the RESA is triggered by the criterion “hot gas temperature higher than 750°C” in the case of an increased hot gas temperature.

5.2.1.2 Partial Load Operation

In the event of a longer partial-load operation, the xenon balance concentration remains below that of the full-load operation. With the lowest operating partial load of 50 % rated power, the reactor operation modes with hot gas temperatures of 700 °C and 600 °C are possible. The redundant reactivity, partially compensated by reflector rods with the highest admissible insertion depth, amounts to approximately 1.8 % /U 5.2-4/.

In the case of the discharge of all the reflector rods under partial load and xenon balance, the possible reactivity increase is higher than the rated value of 1.2 % under full load; yet it lies below the maximum reactivity increase under a full load of 2.5 %. Starting from hot gas temperatures of 700°C, according to the applicant’s parameter variances, lower maximum temperatures of the fuel elements are reached during an adverse event (first RESA start-up failure) than in the above-described adverse event under full load. A simultaneous start-up of the blower does not result in an increase of the maximum fuel element temperature.

Starting from a hot gas temperature of 600 °C, the second RESA criterion “hot gas temperature higher than 750°C “ is triggered later in the event of the reflector rods or KLAK pillar extraction; in contrast with the adverse course of failure under full load, the maximum fuel element temperature can increase to

970 °C at time of the RESA /U 5.2-4/. Taking into account the calculation uncertainties, the maximum rises to 1070 °C.

An even higher maximum fuel element temperature arises on the assumption that the failure occurs after a cold start at approximately 55 % reactor power. In this case, the KLAK pillars are partially filled if in this load point and at the minimum hot gas temperature of 585°C is imputed to a simultaneous extraction of the reflector rods, and a KLAK pillar is assumed, the maximum fuel element temperature of approximately 1030 °C arises at the RESA with regard to the second RESA criterion "hot gas temperature higher than 750 °C," without taking into account calculation uncertainties. /U 5.2-4/. After examining any possible reactivity failures within the total load range from zero load to full load /U 5.2-4), this is the highest value, provided that the second RESA stimulus is considered. Assuming a faulty service of the small pellet switch-off systems during reflector rods extraction, this course of failure may result in higher fuel element temperatures than in the adverse event described in the safety report.

Taking into account the calculations in the security report as well as the document /U 5.2-4/ concerning the extraction of the reflector rods at 50 % without a KLAK pillar defective ride, it is possible to assess that in the event of a KLAK-pillar defective ride, the maximum fuel element temperature at time of the RESA will rise to 1130°C with respect to the calculation uncertainties. By analogy to the calculations under full load, the maximum of the maximum fuel element temperature of 1170°C occurs one hour after a RESA, provided that secondary heat is subsequently discharged via the main heat transportation system.

According to the analysis described in the safety report, the rapid shut-off criterion "neutron flow in medium range higher max. and period lower than 20 s" can be reached in a few seconds. In the event of a presumed failure of this signal, after approximately 35 s, even before any substantial increase in the reactor power (above 5% power), the second rapid shut-off criterion "neutron flow in medium range higher max. and power range not released" is attained. The reactor is shut-off due to falling in of the reflector rods. Due to the RESA-deduced blocking of the KLAK pillars, a further discharge of the KLAK pillar is prevented. No substantial temperature increase occurs. The limitation of the power increase occurs in the cold condition via negative temperature coefficient as shown in additional calculations in /U 5.2-4/. Irrespective of a reactor protection system intervention, the reactor power would be stabilized at a low level. Via the reactor rapid shut-off, this power limitation is eliminated.

A defective reactivity feed in cold/zero load condition as a result of the extraction of all reflector rods and the extraction of a KLAK pillar have also been examined by the applicant. /U 5.2-.4/. In this case, the reactor power would be stabilized at approximately 10%, irrespective of the reactor protection system intervention. For shut-off, both above-named impulse criteria are available. The calculated temperatures lie well below the operating values.

An involuntary reactivity feed, starting from a very low power (approximately 1% reactor power) at the helium temperature considered in the cold start graph, was examined by the applicant. /U 5.2-4/. The highest power increase occurs with the extraction of all reflector rods and the simultaneous extraction of a KLAK pillar. Supposing that the first RESA impulse "neutron flow in medium range higher max. and period lower 20 s" does not take effect, the reactor power stabilizes at nearly 20 % due to the negative temperature coefficients. No further RESA impulse is available in such a case. In our opinion, this condition is acceptable. In our own estimation, the hot gas temperature will stabilize at approximately 400°C within a period of more than 10 minutes. The full-load rated values of the fuel elements and the helium temperatures are not reached.

In the case of an involuntary reactivity feed, starting from a 20% reactor power, the reactor power rises to more than 120% and the second RESA criterion "neutron flow higher than 120 %" becomes effective. Moreover, irrespective of such an impulse, the reactor power stabilizes at such high level that the hot gas temperature rises above 750 °C and the criterion "hot gas temperature higher than 750 °C" becomes effective. This criterion also becomes effective in the case of a higher initial power.

The defective reactivity feed at zero condition and low power is controlled; the design limits are not reached.

5.2.1.4 Start-up Failure

The maximum reactivity feed with a cold undercritical reactor is considered as a start-up failure.

To the cold undercritical reactor can only be supplied the reactivity in that first the reflector rods are extracted from their insertion position into their lowest operating position so that the KLAK extraction is triggered /U 5.2-3/. The KLAK pillars can then be extracted one after the other.

The maximum possible temperature increase in the case of a start-up failure is analyzed in /U 5.2-1/ by the applicant. In this analysis, it is presumed that, simultaneously with the extraction of the KLAK pillars, the reflector rods are extracted from their rated position. The heat discharge from the core and the RESA are not taken into consideration. On these assumptions, the maximum fuel element temperature stabilizes at approximately 1300 °C, while the power generation completely drops again due to negative temperature feedback

Similarly as in the case of the defective reactivity feed in the condition of zero load/cold, the following rapid shut-off criteria are available in the case of a start-up failure for shutting off the reactor:

- "neutron flow in medium range higher max." and "period lower 20 s,"

- "neutron flow in medium range higher max." and "power range not released."

Due to the negative temperature feedback, as well as due to the fact that the second shut-off criterion is triggered below a power of approximately 1 % /U 2.5.5-12/, no substantial temperature
Increase takes place. This is also proved by the calculations in the report (U 5.2-4) of the reactivity failures with a zero load.

In our opinion we have proved that, if the above-mentioned evaluation criteria are taken into account, the start-up failure will be under control.

5.2.1.5 Summary Evaluation

The analyses that are described in this paragraph confirm that, when the reflector rods are released, even if additional failures are taken into consideration, the reactivity lifters and ramps of the absorber cannot cause temperatures that would exceed the limit values of this design. Unacceptable performance excursions do not occur.

The RESA necessary for the limitation of the maximum temperatures of the pressure tank unit is triggered by means of two different stimulation criteria or the performance of the reactor is stabilized at a low level. Even if the first stimulus fails, the reactor is shut off safely assuming that a reflector rod does not fall off.

The temperature limit values of the pressure tank unit are not exceeded. The maximum temperature of the fuel elements remains far below the limit of 1,620°C in all cases, while the fuel element temperature of 1,200°C is slightly exceeded in a few cases. Thus, all possible failures of reactivity resulting from the operation of absorbers can be kept under control taking into account the above-mentioned evaluation criteria.

The investigations described below confirm that the breakdown "release of all reflector rods due to various reasons" is a breakdown that covers all the failure conditions of the control and switching elements.

5.2.2. Erroneous Operation of the Individual Absorbers

5.2.2.1. Erroneous Movements of the Individual Pellet Switching Elements

An erroneous fall of a KLAK pillar in the power operation causes an azimuthally inclined load. Due to a drop of the integral reactor power in the case of a quick complete fall of a KLAK pillar, the high-speed shut-off criterion "negative sliding limit value of the thermally controlled neutron flow higher than 20%/min" is started.

If a side reflector bore is filled slowly and if the control is faultless, no high-speed shut-off criterion is triggered. The input of the negative reactivity due to the fall of the KLAK pillar can be compensated by means of the reactivity reserve of the reflector rods of 1.2% planned for the operation.

In accordance with the applicant's data, an azimuthally inclined load caused by a filled side reflector bore can be recognized by:

- an unusual reflector rod position,
- the message "storage closure of the supply tank open."
- the inclined load control.

Then, the reactor is shut off manually. The temperature distribution caused by the azimuthally inclined load is taken into account in the mechanical design of the side reflector (U 2.5.3-5). In accordance with paragraph 2.5.3.1, the load of the side reflector caused by this is acceptable.

In our opinion, the automatic reactor high-speed shut-off system is not necessary anyway in this case since the ceramic nuclear inserts are designed for azimuthally inclined loads and because the local excess temperatures of hot gas (max. 30 K) can be reduced by the mixing in the core reflector.

The pellet shut-off system is especially required for a compensation of the reactivity before the xenon balance is achieved. To safeguard the symmetric power distribution, the KLAK pillars must be filled with radial symmetry. For this purpose, the applicant must divide the KLAK pillars into groups and provide for group selection in the case of a manual operation (U 2.5.4-7).

We think that the control concept of the pellet shut-off system presented in U 2.5, 4-7 in connection with the planned filling status controls in the pellet reserve tanks is suitable to ensure a symmetric filling of the KLAK pillars. Unsymmetrical filling levels of the KLAK pillars will be prevented by the group control presented by the applicant. Failures in this group control can be recognized via the inclined load control. In our opinion, the long-term operation with unsymmetrical filling levels of the KLAK pillars cannot be assumed.

Reactivity failures resulting from the emptying of a KLAK pillar after a cold start were investigated by the applicant for the whole power part (U 5.2-4). The calculations show that the consequences of the reactivity failures with regard to the power increase and the maximum temperatures due to the "release of all reflector rods" under various conditions are covered.

5.2.2.2 Erroneous Operation of the Individual Reflector Rods

The erroneous operation of the reflector rods should be compensated by the control system in accordance with the design and should be considered part of the normal operation.

The temperature distribution in the side reflector with a fully inserted reflector rod, with the remaining rods used for reactivity compensation drawn up slightly, is dealt with within the framework of the thermodynamic design (U 5.2-2). The local changes in the hot gas temperature do not amount to more than approx. 30 K (U 5.2-2). The additional unacceptable thermal load on the side and core reflectors does not occur. This temperature difference is insignificant after mixing in the core section.

The temperature differences in the case of a complete release of a reflector rod are covered within the framework of the above-mentioned case. The resulting temperature increase is not critical.

In accordance with the calculation of the applicant, if a reflector rod is released from the nominal position at full load without any intervention of the control system, the reactor power increases to max. of 126% within 55 sec. In the case of an assumed failure of the high-speed shut-off criterion “neutron flow higher than 120%,” the reactor power is stabilized between the values of 110% and 120% due to the negative reactivity feedbacks. Because without the interventions of the control system, there is a power balance between the reactor and the steam generator, the intermediate gas temperature rises. The second high-speed shut-off criterion “hot gas temperature higher than 750°C” is achieved after 360 sec. Up to the shut-off the fuel element, the temperature at the core throughput (central core area) rises by approx. 80 K. This means that the temperature increase is less than in the case of the breakdown “erroneous release of all reflector rods”, wherein a greater reactivity supply takes place. The local thermal loads correspond to the above-mentioned case with the functional control system.

The reflector rods can be moved independently in the bores of the side reflector (U 2.5.4-3). The erroneous operation of more reflector rods positioned next to each other lead to smaller loads than in the case of an erroneous operation of all reflector rods due to the smaller reactivity input. The erroneous operation is covered within the framework of this breakdown by the integrated power and temperature processes. The reflector rods are prevented from being inserted too deeply (for example, due to an erroneous emptying of KLAK pillars) by the control rod operation limitation via a shutdown of the KLAK conveyance (U 5.2-3 and U 2.5.4-7).

In the case of a high-speed shut-off of the reactor, the reflector rods fall about 1 m below the core center. However, the bores of the reflector rods in the side reflector reach under the lower edge of the core. If the drive unit of a reflector rod fails (the round link chain is also a part of this unit), the rod can fall from its desired position.

The applicant has investigated such a fall of a reflector rod from its nominal position under full load (U 5.2-4). In accordance with this investigation, the reactor power sinks quickly after the fall of the reflector rod, and due to the negative reactivity feedbacks, it is stabilized at values of approx. 85% after 200 sec.

In this case no intervention of the control system is taken into account. In accordance with the investigations of the applicant (U 5.2-4), the fall of a reflector rod has the worst effects if the rods are standing on the lower entrance limit and a group of the KLAK pillars is filled partly in the xenon-free core. The fall of a rod from this position causes positive reactivity input. To avoid positive reactivity input if a reactivity rod falls, the applicant is planning to install a barrier that will prevent the reflector rods from being inserted too deeply when the KLAK pillars are filled only in part. The acceptable insertion depth is determined in such a way that no positive reactivity input occurs in the case of a fall of a reflector rod. In accordance with the calculations made so far, the insertion limit should be positioned near the nominal operational position of the rod (at the depth of about 2.5 m).

In the case of a pressure relief breakdown, first the upper part of the reflector rod bores is relieved. The resulting pressure difference over the reflector rods is not sufficient to prevent the rods from falling. One or more reflector rods are prevented from being driven out by means of the structural design.

5.2.2.3 Summary Evaluation

The reactivity breakdowns resulting from erroneous movements of the individual absorbers are under control. The loads of the plant are not greater than in the case of the erroneous operations of the reflector rods or pellet shut-off units in this case.

5.2.3 Erroneous Start-up of the Primary Circuit Blower

In the case of an erroneous start-up of the primary circuit blower, the hot gas temperature drops and the power of the reactor increases.

In the safety report, the applicant describes the consequences of the erroneous start-up under the lowest operation partial load of 50%. The assumption is that the blower shall start up from 50% to 105% of the nominal throughput in 14 sec. In this way, the maximum throughput is achieved. During the start-up of the blower, the high-speed shut-off criterion "volume ratio (primary to secondary side) higher than 1.3" intervenes. Assuming that this stimulus fails after about 15 sec., the high-speed shut-off criterion "cold gas temperature higher than 280°C" intervenes.

Because during the erroneous start-up of the primary circuit blower, the hot gas temperature drops and the second high-speed shut-off criterion guarantees a sufficient interval to the design temperature of the pressure tank unit of 350°C, we think that unacceptable component loads cannot be assumed.

An erroneous start-up of the primary circuit blower and a simultaneous release of the reflector rods are only possible as a result of a control failure. The applicant has investigated this breakdown (U 5.2-4).

With a partial load of 50%, the reflector rods absorb a reactivity of about 1.8 % (1.2 % control reserve and 0.6% to compensate for the xenon and temperature effect). From this condition of the reactor with a 50% partial load, the blower starts up and within 14 sec. it reaches from 50% to 103% of the nominal throughput and, at the same time, the reflector rods retract. In the analysis is maintained a constant temperature of 250°C and, in this way, the reaching of the second high-speed shut-off criterion is delayed.

The analysis shows a fast increase in the reactor power. Within the start-up time of the blower is reached the high-speed shutdown criterion "ratio of the throughput of the primary and secondary circuit higher than 1.3." After about 20 sec is reached another shut-off criterion, the "neutron flow of higher than 120%." The increase of the maximum fuel element temperature is not significant due to the timely shut-off.

The temperature transients are less than in the case "release of the reflector rods" under full load. This breakdown does not cause any unacceptable loads.

When the hot reactor is shut off and the released heat is discharged by means of the surface cooler, in the case of the erroneous start-up of the primary circuit blower, hot gas with a high temperature can be supplied to the steam generator. If the start-up takes several hours after a shut-off from full load – the helium temperature in the reactor pressure vessel is then high – the design temperatures for the steam generator components can be exceeded. Therefore, it must be proved that exceeding these design temperatures for a short period of time is acceptable. Otherwise the erroneous start-up of the blower with high helium temperatures must be avoided by means of a shut-off.

5.2.4 Reduction of the Cold Gas Temperature

If the cold gas temperature drops, the reactivity is delivered to the reactor. If this reactivity is not compensated by means of the control system (inserting the reflector rods) or by means of a manual intervention (filling the KLAK pillars), the power of the reactor is increased.

The applicant investigated the consequences of a cold gas temperature drop (for example, due to failures of the feed water supply) under full load in a parameter study (U 5.2-4). Within 30 sec. the cold gas temperature is brought from its nominal value of 250 °C to a new stationary value. The lowest cold gas temperature in the parameter study is 150 °C. This temperature is maintained in the case of a failure of the back-up steaming by means of the LP-preheating; if the LP-preheating is transmitted, a slightly higher temperature is achieved through the back-up steaming (U 2.6.2-5).

The calculation of the stationary operation conditions of the steam generator (5.2-4) under full load shows a difference of 66°C between the feed water temperature and the cold gas temperature. This means that a feed water temperature lower than 100°C would be required to achieve the cold gas temperature of 150°C. This is a value that is not realistic in normal operation. It could only be achieved in the case of a failure of any corresponding pre-steaming. The calculations of the partial-load operation conditions and the cold start show minimum cold gas temperatures of 178°C (U 2.6.4-1).

We consider the parameter study performed by the applicant as conservative with respect to the high input of reactivity, as well as with respect to the gradient and the minimum value of the cold gas temperature of 150°C. The parameter study shows that two physically different high-speed shut-off criteria are available if the RESA is required due to the power balance between the reactor and the steam generator. In the case of a shut-off due to the second criterion, the maximum fuel temperature rises to 950°C if all inaccuracies in the calculations are taken into account.

The effects of the maximum drop of the cold gas temperature can be classified among those that occur during the "release of all reflector rods." The limit values for the design have not been exceeded.

In the case of a break of the fresh steam supply pipe, the drop of the cold gas temperature is limited by means of throttling in the steam generator. The temperature drop is slight (see paragraph 5.4.2). The reactivity feedback are less than in the case of the above-mentioned failure.

The failures on the secondary side that cause the cold gas temperature to drop are kept under control if the above-mentioned evaluation criteria are taken into account.

5.2.5 Water Penetration into the Primary Circuit

Because of a break of a heat exchanger pipe, a max.of 600Kg of water flow into the primary circuit (see chapter 5.4.3). Under the assumption that this water mass is distributed evenly in the primary circuit, the applicant has calculated an increase of reactivity amounting to 0.4%.

The penetration of water from the break point to the primary circuit takes place within an interval of at least 40 sec. With the conservative assumption that the reactivity input of 0.4 % proceeds in a linear manner throughout this period of time, then a reactivity input rate results, which corresponds to the rate when all reflector rods are released under the full load (U 2.5.5-16). However, the amount of the reactivity input rate has no significant influence on the reactivity breakdown process, as is shown in the parameter variation (U 5.2-1).

In the case of a penetration water pipe, the total supplied reactivity is less than in the case of a release of all reflector rods. Therefore, the thermal loads on the components due to the power increase after a break are overridden by the design breakdown "release of all reflector rods."

The reactivity supplied because of water penetration is taken into consideration in the shut-off balance (U 2.5.5-14).

We can confirm that the reactivity increase due to a penetration of water into the primary circuit is controlled with respect to the above-mentioned evaluation criteria.

5.2.6 Densification of the Fuel Element Filling During an Earthquake

The vibrations during an earthquake can increase the filling factor of the core and feed more reactivity in a short period of time.

The applicant mentions that, in the case of an earthquake acceleration of 0.5 g, the filling factor of 61% can increase to 61.4 % within 6 sec. The assessment of the vibration experiments with the SAMSON plant (U 2.5.5-17) has shown a relative increase of the filling factor of 1.0065 with the acceleration of 0.5 g within 6 sec. By using this figure one can calculate that the reactivity increase is 0.125%. This fast reactivity growth causes the reactor power to rise from 100% to about 150% within 20 sec.

Regardless of the high-speed shut-off criteria "period shorter than 20 sec." and "neutron flow higher than 120%," the reactor power would be stabilized at the level of approx. 110% after about 200 s. because of the inherent feedback properties. However, the second RESA starting criterion would be initiated after 4 sec., which means that the above-mentioned maximum value of the reactor power of 150% would not be reached.

We have tested the amount of the reactivity input. The experiments with pellet heaps (L 135) have shown that the densest random pellet pack occurring during the first filling of a tank becomes loose due to the rolling movements of the pellets. After about 40% of the tank volume is rolled over, the filling factor reaches the saturation value. For the geometry of the HTR pellet filling, the saturation value is calculated as 60.8%. This figure confirms the filling factor used by the applicant. Decisive for the magnitude of the reactivity supply is, however, not only the absolute value of the filling factor in operation, but its relative increase.

The results of the vibration experiments show that the relative growth of the filling factor achieves a maximum value only after 1 min and the maximum value depends on the acceleration of the vibrations. At the beginning of the vibrations, the relative filling factor first rises steeply, but the growth in the first few seconds has not been accurately proved in experiments. To take this inaccuracy into consideration in a conservative manner, we have used as the basis for the relative growth in the first 6 sec. the value of 1.01. This means that the filling factor rises to 61.6%.

Since the relativity growth depends on the increase of the filling factor in an approximately linear manner, the reactivity input of 0.19% is derived from the relative growth of 1.01. With the simplifying assumption that the maximum reactor power in the area to be observed depends approximately linearly from the reactivity input, we detected a maximum reactor power as a consequence of the vibrations of an assumed earthquake of under 250 % when the short-term RESA stimulus is not taken into consideration.

The testing stimulus of 0.5 g used during the course of the trials performed to provide evidence must be compared to the maximum of the earthquake stimulus at the installation location of the core container. This figure can be determined as the result of the dynamic earthquake calculations with various parameter variants or it can be read as the solid body acceleration from the layer response spectra for the installation location (U 2.6.6-2), (U 5.2-5). The comparison shows that the stimulus acceleration of the HTR core due to the assumed seismic load is, in any case, significantly less (about 0.28 g) than the testing stimulus of 0.5 g, in accordance with the location assumptions.

Another influencing factor for the transferability of the testing results is the geometry of the core design. The tests for the pellet heap densification were carried out for the core in the THTR scale (H/W = approx. 1) (U 2.5.5-17, U 2.5.5-18). The HTR module core is different from the testing model with respect to the core geometry (H/W = about 3) and the structure of the core units. It can be subjected to higher acceleration and longer oscillations due to oscillation excess, so that the impulse acceleration and duration itself may not be sufficient as the evaluation criterion for the densification as in the case of a solid body. The applicant takes this effect into consideration by using the interval between the testing acceleration (0.5 g) and the calculated impulse acceleration at the HTR module core. However, we consider that is necessary that during the implementation planning the transfer conditions for using the testing results for the HTR module be dealt with in a more detailed way and that corresponding analytic or experimental security measures be taken.

The breakdown "erroneous release of reflector rods under full load" proves that a fast power increase, even over 200%, is still under control if the RESA only intervenes through the high-speed shut-off criterion "hot gas temperature higher than 750°C." The temperature increase caused by the fuel element densification in the coated particles and the fuel matrix lies far below the destruction limits specified in the KIWI-TNT experiments (L 138).

5.3 Breakdowns of the Heat Discharge without Cooling Agent Loss

In section 5.1 of this assessment are mentioned events that can cause breakdowns of the heat discharge without a loss of cooling agent. They comprise the following:

- an interruption of the primary cooling agent throughput,
- breakdowns in fresh steam removal,
- breakdowns in the feed water supply,
- a failure of the own consumption supply.

These breakdowns cause deviations of the thermodynamic conditions from the normal operation. These deviations are recognized by the limitation and reactor protection devices and necessary protective actions are taken.

A failure of operation controls is generally expected. It has been proved that all breakdowns can be controlled by the reactor protection system. In our opinion, the extent of the functions of the operation control systems in critical failure conditions can only be tested in a detailed way within the development design of the secondary end. However, this is not necessary for the concept evaluation, since the boundary conditions mentioned in the evaluation criteria are not affected.

Using the superior rules and regulations quoted in chapter 5.1 we have summarized the following basic evaluation criteria:
The reactor core and the devices, which are important from the technological safety point of view, must be designed in such a way that they do not exceed the boundary conditions specified for breakdowns. This is the case when the fuel element temperatures remain limited to the values of about 1,620°C and the load of important parts of the system from the technological safety point of view

remains below the design values. The design pressure of the pressure tank unit (DBE) amounts to 70 bar. It lies above the triggering pressure of the safety valve. For the course of a breakdown, the pressure processes in DBE are of little significance.

The most important design temperatures to be maintained are:

- the fuel elements	1,620°C
- the pressure vessel unit (RDB and DE coat, (connecting pressure vessel)	350°C
- the core container	500°C
- the bottom structure	500°C
- the hot gas line	350°C
- the reflector rods	650°C
- the gas pipe of the hot gas line	900°C
- the team generator units	
- the overheater pipe, gas side	750°C
- the overheater pipes, steam side	570°C
- the evaporator pipes	500°C
- the fresh steam pipe plate	570°C
- the feed water pipe plate	350°C
- the fresh steam and feed water system	
- the fresh steam nozzle	540°C
- the fresh steam line	540°C
- the feed water nozzle	540°C
- the feed water line	350°C
- the reactor body (concrete)	150°C

The applicant analyzes breakdowns of the heat discharge without cooling agent loss by means of the DEVIL program (U 5.3-5). This program describes the transient behavior of the steam generator during a failure of the feed water and fresh steam. It calculates the load of the steam generator and the resulting effects on the reactor core.

The steam generator is described in a mathematical-physical way using a representative pipe, the description of the flow is one-dimensional. At the primary side is allowed a backflow, the secondary side is described as a compressible homogenous two-phase flow. The steam generator is divided into about 50 axial zones.

The calculation of the pressure profile at the water side takes into account static as well as dynamic pressure fractions. The pressure drop in the case of the two-phase flow is described in compliance with Martinelli and Nelson. The pressure at the feed water inlet consistent with the prescribed fresh steam pressure (DE outlet) is determined in an iterative way.

The heat transfer to the He side is calculated in compliance with Grimison.

The calculation of the heat transfer at the water side is carried out in compliance with the following ratios:

- the subcooled area according to Dittus, Boelter,
- the boiling area (subcooled and saturated) according to Thom,
- the film boiling according to Dougall, Rohsenow,
- the overheated area according to Dittus, Boelter

The DEVIL program was verified experimentally by means of an additional calculation of the measurements of the He-heated steam generator of the component testing plant (U 5.3-6). During the measurements, the H_2O throughput was changed in the range of about 1:1.5. The pressure drop on the testing section and the medium outlet temperatures were checked.

The comparison of the calculation with the measurements shows a satisfactory coincidence. The possible deviations of the calculated values from the measured values have been explained by the applicant, first as a result of a missing description of the structural details of the KVK steam generator in the DEVIL calculation model, and second as a result of an underestimation of the pipe friction influences in the pressure loss calculation. This explanation seems plausible. As the experience shows, such influence parameters need to be adapted in the cases of the design of real technological plants by means of calculation models and they do not question the general capability of the program.

The ratios used in the program to describe the stream and heat transfer processes are well known from relevant specialized literature and have been proven. This is why we find the DEVIL program generally suitable for the purposes of simulating the behavior of the HTR module steam generator.

5.3.1 Interruption of the Primary Cooling Agent Throughput

An interruption of the primary cooling agent throughput is caused by a failure of the primary circuit blower or by a faulty closure of the blower valve. The secondary heat discharge takes place via the area cooler.

failure of the primary circuit blower can be caused by a breakdown of the current or oil supply or in the oil heating or due to a blocked or broken blower shaft. In a very short period of time (a few seconds), all these failures lead to an interruption of the primary cooling agent throughput. This interruption is recognized by the reactor protection system by means of the following triggering criteria:

- the throughput ratio (primary to secondary side) less/equal 0.75.
- the negative sliding boundary value of the thermally controlled neutron flow less than /equal to about 20%/min.

The 1st initiation criterion (throughput ratio) is effective after about 2 sec. (U 4.2-1). The applicant has assumed a failure of the 1st initiation criterion in the analyses. After that, the 2nd initiation criterion (neutron flow) triggers the reactor protective actions after approx. 23 sec. (inserting the reflector rods, switching off the blower, closing the secondary circuit). The primary cooling agent flow rate drops gradually with a blower half-value time of 4 sec. The blower valve closes 3 sec. after the start of the reactor protective actions and after another 15 sec. it interrupts the primary cooling agent throughput.

The fresh steam throughput is reduced due to the low power supply resulting from the interruption of the primary cooling agent throughput. When the protective action becomes effective, the feed water throughput is shut-off immediately due to the closed feed water valve; as a result of this, the fresh steam throughput drops quickly. About 30 sec. after the start of the reactor protective action, no more fresh steam throughput is available as a result of the closed FS valve (U 4.2-1).

Due to the interruption of the primary cooling agent throughput, the intermediate hot gas temperature rises slightly (about 4°C). The structural design of the gas system inside the pressure vessel unit (RD3 and DE-jacket, connecting pressure vessel) makes it impossible for the hot gas to come into contact with the pressure vessel walls. (U 2.6.2-3). The long-term temperature processes of the interior unit of the reactor pressure vessel do not differ considerably from the RESA manual processes and the long-term heat withdrawal via the surface cooler. In this case, the maximum temperature of the fuel elements lies significantly below 1,200°C. Just about 5% of the fuel elements achieve temperatures above 1,000°C. The maximum temperatures of the ceramic inserts remain approximately below 1,000°C. The core container reaches a maximum temperature of about 400°C and the pressure vessel reaches a maximum temperature of about 320°C.

Due to the rapid throughput drop at the primary side from 530°C to 360°C, the fresh steam temperature drops. The cold gas temperature drops from 250°C and it comes near the constant feed water temperature of 170°C. The temperature processes, with the exception of the fresh steam temperature, differ just insignificantly from the corresponding processes of the RESA started manually and the long-term secondary heat discharge via the surface cooler. As a consequence of the premature abrupt backflow of the feed water in the case of the RESA, the fresh steam temperature rises temporarily.

The long-term temperature process of the inserts in the steam generator pressure vessel and in the fresh steam and feed water system, is characterized by falling temperatures resulting from the lack of power production at the primary side.

If the primary cooling agent throughput is interrupted, the secondary heat removal is carried out at the operation pressure by means of a heat transfer and radiation via the surface cooler. This is why the maximum fuel element temperature remains under 1,200°C.

The design temperatures of the metallic parts of the primary and secondary circuit are maintained.

If it is assumed that the blower valve having only such an operational function does not close, then the primary cooling agent throughput is shut-off later. The additional power supply from the primary side to the secondary side is negligible in comparison to the total power supply during this failure. A natural circulation starts in the primary circuit.

The maximum gas temperatures in the primary circuit in the area of the metallic components remain under 400°C on a long-term basis.(U 4.2-2). Here, it can be assumed conservatively that the steam generator evaporates. Even if a single failure and maintenance of the surface cooler is assumed (3 x 100%), this does not impair the additional heat discharge. Even on a long-term basis, the temperature of the core containers remains below the maximum considered value of 500°C that occurs in a case of a pressure relief failure.

5.3.2. Failures of the Fresh Steam Removal

Failures of fresh steam removal can occur due to the following events:

- an erroneous increase of the fresh steam removal due to
  - an erroneous opening of a turbine by opening an inlet valve,
  - an erroneous opening of a fresh steam valve,
  - an erroneous opening of a fresh steam safety valve,

- an erroneous drop of the fresh steam removal due to
  - the high-speed shut-off of the turbine
  - the response of the capacitor protection
  - a closure of a fresh steam bypass station (FD reduction station),

- an erroneous closure of a fresh steam valve

Failures of the fresh steam removal resulting from fresh steam line breaks are dealt with in Chapter 5.4.2. The errors concerning the process steam removal are covered by the failures described in this chapter.

In the case of an erroneous increase of fresh steam removal due to the above-mentioned failures, both the turbines are throttled by the control system. In cases when the fresh steam pressure cannot be maintained at the correct level due to a failure of the fresh steam pressure control, the power of the plant drops.

A failure of the fresh steam pressure control causes an increase of the feed water throughput. The increased steam generator cooling resulting from this fact leads to increased power in the reactor, which is either compensated by the control system or leads to a high-speed shut-off of the reactor as a result of the increased hot gas temperature.

The effects of the faulty increase of the removal of fresh steam to the steam generator and the primary circuit can be compared to those after an erroneous increase of feed water throughput (compare Chapter 5.3.3).

In the case of an erroneous closure of the fresh steam valve, the pressure in the steam generator rises up to the release pressure of the fresh steam safety valve, which opens as a result of this. With the increased fresh steam pressure, the feed water valves are closed at the same time and the RESA is initiated through the starting criterion "throughput ratio of the primary/secondary circuit." In this way, the long-term shut-off of the safety valve is avoided. There are no significant effects on the primary circuit since the heat removal at the secondary side is not interrupted.

If the fresh steam safety valve is closed due to a single fault, the pressure in the DE rises until the steam generator relief valves are activated (U 2.9-6). As a result of this, the RESA is stimulated again through the initiating criterion "negative sliding boundary value of the fresh steam pressure." The development design should prove that the acceptable loads of DE are not exceeded.

In the case of all failures of fresh steam removal, the additional heat removal is secured by means of a heat transfer, heat radiation, and convection via the surface cooler or the central heat transfer system. The reactor is under the operation pressure. The maximum FE-temperature remains below 1,200°C. The additional heat removal via the steam generator is insignificant due to the design and low energy supply to the steam generator.

5.3.3 Failures in the Feed Water Supply

The failures in the feed water supply include the total failure of the feed water throughput as a consequence of a failure of feed water pumps or due to an erroneous closure of the feed water valves. The other failures can result from an increase or drop of the feed water throughput caused by the control system.

A failure in the feed water supply resulting from a break of the feed water line is dealt with in Chapter 5.4.2.

An immediate consequence of the total feed water failure is a rapid throughput drop at the secondary side leading to an increase of the cold gas temperature.

The reactor protective system is able to recognize the failure by the following triggering criteria:

- a throughput ratio (primary to secondary side) higher than/equal to 1.3,
- a cold gas temperature higher than/equal to about 280°C.

The first triggering criterion (throughput ratio) becomes effective after 1-2 sec. The applicant has suggested the failure of the 1st criterion (throughput ratio) in their analyses. The 2nd triggering criterion (cold gas temperature) is triggered at the latest after 36 sec. The reactor protection measures (inserting the reflector rods, switching off the blower, shutting off the secondary circuit) (U 4.2-1).

The primary cooling agent throughput remains constant until the triggering moment of the reactor protective actions. 3 sec. after the RESA intervention, the blower valve closes and, after another 15 sec., it is closed. If the blower valve does not close, the failure is not worsened by this fact.

The fresh steam throughput is reduced due to lacking feed water and is interrupted after an accomplished intervention of the RESA via a closure of the FD closing valves.

Because of the unbalance between the production and discharge of the power, the hot gas temperature rises slightly (about 4°C), and the fresh steam temperature comes closer to the hot gas temperature of 700°C. The cold gas temperature rises to approx. 350°C. The feed water temperature remains constant. The long-term temperature development of the primary circuit components can be compared to the development after a RESA intervention and to the long-term heat removal via the surface cooler (see Chapter 5.3.1).

In the case of a failure-dependent start-up of the feed water pumps, the power produced in the core rises. The applicant has analyzed this case with a feed water throughput increased by a jump-like throughput of 130% and a blower speed of 103%. At the same time the applicant suppressed the occurring RESA impulses to be able to determine their chronology.

After about 1.55 hours, the hot gas temperature reaches the RESA boundary value of 750°C. A few moments later, that is, 1.6 hour after the beginning of the failure, the power reaches the RESA boundary value "neutron flow ≥ 120%." The fresh steam temperature falls by approximately 130°C for a short time. After this, it starts to rise slowly again.

Due to the fact that during these breakdowns the reactor is under the operation pressure and that the secondary heat is removed through the heat line, the radiation and convection via the surface cooler, the maximum fuel element temperature remains below 1,200°C.

During these breakdowns, the fresh steam temperature comes nearer to the hot gas temperature of about 700°C. This is why the design temperatures of the fresh steam pipe plate (570°C), the fresh steam nozzle (540°C), and the fresh steam line could be exceeded. In correspondence with the intention statement of the applicant, this should be avoided by means of a limiting device that initiates a temperature drop by means of the reflector rods in case the required value of the fresh steam temperature is exceeded and shuts off the reactor if necessary. However, corresponding details and analyses have not been submitted yet. They must be presented within the development design. We do not consider this aspect to be of great significance for the concept.

Just the function of the surface cooler that is already in operation is required to control the breakdown. The individual failures and maintenance are controlled via the redundancy of the surface cooler (3x100%).

In the case of this breakdown, the primary cooling agent throughput is also shut off later when the blower valve with the operational function remains open. The resulting additional power supply from the primary side to the secondary side is insignificant when compared to the total energy supply during this breakdown. The maximum gas temperatures in the primary circuit in the area of metallic components remain below 400°C for a long time thanks to the natural circulation (U 4.2-2). If the redundancy of the surface cooler cannot be used due to maintenance, the heat removal is not impaired. The loads on the core container are covered by the loads occurring in the case of a pressure discharge breakdown.

The analysis of the failures of feed water throughput (U 5.3-4) is carried out under the covered viewpoints. At the same time, the maximum increase and drop of the feed water throughput due to faulty control are assumed. A comparison of the results of these analyses with the design temperatures of the components that are important from the point of view of the safety shows that there are no loads of the components that have not been dealt with by the design.

5.3.4 Failure of the Own Consumption Supply

A failure of own consumption (stand-by power supply) supply occurs in the case of a failure of the connecting mains and a simultaneous failure of the own consumption supply by means of the turbo-generator. The secondary heat is discharged via the surface cooler. Operation leaks from the primary circuit to the secondary circuit are excluded because of pressure in the primary circuit is lower than in the secondary circuit.

The failure of the own consumption supply leads to a failure of the primary circuit blower, the feed water pumps, and the cooling circuits of the surface cooling system that are not supplied with stand-by power as well as to a drop of the reflector rods. The blower valve is closed if a stand-by power supply is available. From the technological safety point of view, the closing of the blower valve is not required. This event is recognized by the reactor protection systems through the following starting criteria:

- the throughput ratio (primary to secondary side) higher than/equal to 1.3,
- the negative sliding boundary value of the thermally controlled neutron flow is higher than/equal to approx. 20%/min.

The 1st starting criterion (throughput ratio) becomes effective after about 1-2 sec. and the 2nd starting criterion (neutron flow) after about 23 sec.

The applicant has analyzed the system behavior after the triggering of the 2nd criterion. The neutron flow as a triggering event already leads to actions that can be compared to those of the RESA. The most important difference with the RESA is that in the case of this event, the feed water pumps are emptied, which results in a faster return of the feed water than in the case of a closing the secondary circuit. Therefore, the consequences of this event, that is, the temperatures in the primary and secondary system as recognized by the 2nd starting criterion, can be compared to those in the case of the 1st criterion which become effective. When the 2nd criterion becomes effective, the reactor protection measures (inserting the rods, switching off the blower, DE closure) are triggered, unless they have already occurred because of initial event.

After a failure of the own consumption supply, the primary cooling agent throughput drops. Due to the fact that the blower valve can only be closed when the stand-by supply has been started, the primary agent throughput is stopped about 30 sec. later than in the case of the RESA manual intervention.

The fresh steam throughput also drops since the feed water supply is immediately interrupted due to the interrupted feed water pump.

Because of the shut-off primary circuit blower, the intermediate hot gas temperature rises slowly (about 4°C). Due to the sudden drop in the feed water supply, the fresh steam temperature drops from 530°C to about 650°C.

The beginning the cold gas temperature initially remains approximately constant and then drops from about 250°C to about 220°C after the standstill of the primary cooling agent blower. The feed water temperature remains constant.

During the own consumption supply failure, the long-term temperature behavior can be compared to a manual intervention of the RESA. The maximum core temperature lies below 1,200°C. Due to the increase of the fresh steam temperature to about 650°C, the design temperatures of the fresh steam pipe plate (570°C), the fresh steam nozzle (540°C), and the fresh steam line can be exceeded. This case has already been discussed in the previous chapter. The temperatures in the steam generator drop on a long-term basis due to missing power supply.

The surface cooler consists of three lines (3x100%), of which two systems are connected to the stand-by supply. In the case of an individual fault and maintenance of both these systems, no energy is available for the operation of the surface cooler. In this case, the reactor is under pressure. Due to the natural convection in the core, the head is distributed faster to the RDB and to the surface coolers than in the case of a pressure relief failure.

The applicant distinguishes the following cases:

- a short-term failure of the own consumption supply (up to 2 hours),
- a longer-term failure of the own consumption supply (up to 15 hours),
- a long-term failure of the own consumption supply (longer than 15 hours),

always under the boundary condition of the lack of non-availability of the two stand-by power generators.

In the case of a short-time failure after the return of the power supply, the plant is started up again via the main temperature drain within the standard operation. The design limits were not reached.

In the case of a long-term failure, the plant finds itself in a safe condition, that is, the reflector rods are inserted, the blower is switched off, and the DE is closed. After 2 hours, the control station is no longer supplied with current and is therefore out of operation. Due to the failure status directed to initiate the reactor protective system, the primary circuit is shut off and the DE relief is started. During the warm-up phase of the core, the primary circuit safety valve can respond. If the power supply is renewed within 15 hour, the surface cooler is set into operation and the plant is started again via the start-up circuit.

The applicant has performed extensive analyses concerning this case using the TAC 2D program (U 2.10-1). These analyses show that in the spatially limited reactor body (the primary cell concrete) the design temperature of 150°C is reached in 15 hours. The maximum pressure vessel temperature rises to 310°C within this period; the surface cooler reaches 220°C. The maximum fuel element temperature does not depend on the function of the surface cooler and remains below 1,200°C. The maximum temperature of the reflector rods lies above the design temperature of 650°C. We have expressed our opinion about this case in Chapter 2.5.4.2.

The TAC 2D program is well known; it is generally recognized as the program for calculating the thermodynamic design of high temperature reactors (see Chapter 5.4.1).

As a conclusion, we determined that where there is a total failure of the surface cooler with the reactor under pressure, no unacceptable loads of the reactor pressure vessel including the inserts, the reactor body, and the surface cooler occur within a period of 15 hours.

No long-term failure of the own consumption supply for more than 15 hours is not assumed within the design range. In this case, the external supply of the surface cooler (fire emergency connections) and a prepared mobile stand-by generator ensure that the design limits of the reactor pressure vessel are not exceeded (U 2.10-1).

5.3.5 Summarizing Evaluation of the Failures of Heat Removal without Cooling Agent Loss.

When all of the above-described failures of the heat removal without cooling agent loss occur, the secondary heat is removed via heat transfer, heat radiation and convection via the surface coolers or via the main heat transfer system. The reactor remains under operating pressure. The maximum FE temperature remains below 1,200°C. The secondary heat removal via the steam generator is insignificant due to the layout and low power input of the steam generator.

The long-term temperature behavior of the primary circuit components is similar to the behavior in the case of a RESA manual intervention with a long-term heat removal via the surface coolers. The differences are negligible and have no safety-technological significance.

The differences concerning the triggering points of the reactor protective actions have no considerable effect on the long-term behavior (hours to days) of the plant due to the HTR specific properties – long delay periods resulting from a comparably high heat capacity of the fuel elements and the ceramic inserts. The extent of how much of a deteriorating effect the control systems can have on the failure conditions can, according to our opinion, only be tested within the development design of the secondary circuit.

During the breakdowns "total failure of the feed water throughput" and "failure of the own consumption plant," the design temperatures of the individual components can be exceeded. In correspondence with an intention statement of the applicant they should be prevented via a limiting device (U 2.6.7-1).

In the case of an assumed total failure of the surface cooler within the back-up power supply, it is not necessary to take any measures within the first 15 hours with regard to the relatively low power production and slow temperature development of the core.

The limit temperature of the fuel elements is maintained without the necessity of using active components. In this way, a massive fission product emission from the core is excluded.

5.4 Failures with Cooling Agent Loss

5.4.1 Leaks and Breaks at the Primary Side

In accordance with the description mentioned in Chapter 5.1, leaks and breaks of the pipes of the primary cooling agent are assumed. Within the breakdown analysis are investigated the following cases:

- A break of a large connection pipe,
- A break of a small pipe and small leaks,
- Breaks and leaks of the pipes which conduct primary cooling agent outside the reactor building.

Breaks and leaks of the pipes conducting the primary cooling agent can cause a partial or total pressure loss in the primary system. The pipelines connected to the pressure vessel unit are:

- 1 fuel element conveying pipe		DN 65
- 2 fuel element removal pipes		DN 65
- 2 fuel element break removal pipes		DN 125
- 1fuel element gas backflow pipes		DN 50
- 3 carrier gas lines for the small pellet shut-off system		DN 65
- 1 pressure balancing pipe		DN 65
- 3 oil pipes for the primary circuit blower		DN 50
- 2 connecting pipes for the helium purification plant		DN 65
- Measuring pipes	≤	DN 10

With the exception of the fuel element break removal pipe, all the pipes transporting primary cooling agent have a diameter of DN 65 or less. Both the fuel element break removal pipes have a diameter of DN 125. However, the possible free cross section is reduced by means of a bore hole in the partitioning plate in such a way that the leakage amount is at the most as large as in the case of a break of a DN 65 pipe

Except for the pressure balancing pipe, all the pipes with a greater diameter than DN 50 have primary circuit closing valves. Breaks or leaks that cause a pressure drop of 180 mbar/min or more trigger a closure of the primary circuit controlled by the reactor protection system. If a failure of automatic shut-off of the primary circuit due to an individual breakdown is assumed, then the corresponding shut-off valve can be closed manually and the radioactive material is prevented from escaping to the environment.

Chapter 5.4.1.1 deals with a DF break of a connecting pipe with a nominal size of DN 65 immediately after the pressure vessel unit. In the case of this break, the greatest flow of leaking mass results in the beginning with the highest pressure drop speed of about 0.8 bar/s. This condition also covers all other leaks and breaks to which cannot be blocked. We have expressed our opinion in Chapter 2.8.with respect to the effects of medium-sized holes on the pressure grading and the directed ventilation necessary for monitoring the hand-over of activities.

From the high level rules and regulations described in Chapter 5.1 we have summarized the following basic evaluation criterion:

The reactor core and the important plant parts and systems must be designed so that, from the safety-technological point of view, the loads occurring during failures always remain within acceptable limits.

The most important design temperatures to be maintained are:

- Fuel elements	1, 620°C
- Pressure vessel unit (RDB and DE coat, connecting pressure vessel)	350°C
- Core container	500°C
- Bottom structure	500°C
- Hot gas line	350°C
- Reflector rods	650°C

- Gas pipe of the hot gas system 900°C

We express our opinion with respect to the maintenance of these temperatures in this Chapter. The remaining load effects of the cases with cooling agent loss are discussed in Chapter 5.4.4.

5.4.1.1. Break of a Large Connecting Pipeline

In the case of a break of a large connecting pipeline at the reactor pressure vessel (RDB), a total pressure loss of the primary system up the balance with the ambient pressure occurs within a few minutes. During this relief phase, the throughput in the reactor core drops in correspondence with the dropping primary pressure, while the power of the reactor until the high-speed shut-off (RESA) at first remains stable and then falls due to the temperature increase and negative temperature coefficients. The failure is recognized via the criterion "negative sliding boundary value of the primary pressure ≥ 180 mbar/min," immediately after it occurs, and it results in a RESA within 10 s. In the case wherein this criterion fails, the RESA is triggered after about 30 s via the second triggering criterion "negative sliding boundary value of the controlled neutron flow ≥ 20%/min" (U 5.4-1).

After the pressure loss, a long-term core heating occurs in the depressurized primary circuit. Since the natural convection processes in the core are low because of the low helium density, the heat is removed in the core mainly via heat transfer and heat radiation. A further heat removal to the surface cooler as a heat drain is ensured via heat transfer in the ceramic and metallic structures in the RBD as well as via heat radiation and natural convection in the gas-filled cavities within (between coal stone, the core container, and the RBD) and outside the RBD up to the surface cooler. The temperature development in the core and the structures is influenced by the temporary process of power produced after decay and the high heat capacity of the core and the ceramic inserts in the RBD as well as the heat resistances between the core and the surface cooler. The pressure loss breakdown (DES) described in this chapter causes further breakdowns which are discussed in this evaluation since, because of the following increase of the core temperature, the highest temperatures of the core and the RDB inserts (shut-off rods, core container, RDB wall) are achieved. and should be seen as conceptual case for these parts.

Analysis Method

To be able to analyze the temperature development in the case of a pressure loss breakdown (DES), the applicant as well as we used the TAC2D (L 139) computing program, but with special modifications in individual cases. This program is used as a framework program for solving the two-dimensional non-stationary heat transfer equation. The framework is completed with a model of the analyzed system and with various sub-programs used to calculate the material values for the various material zones of the system.

The model of the plant presents the geometry of the system with respect to the various materials and is characterized via a suitable grid network and the meshes resulting therefrom (local discrimination). The specifications of the material values are determined by means of various sub-programs that must be obtained by the program user. In this way, various dependencies from the time, the temperature, the geometry, the radiation, and the like can be taken into account in a flexible way.

While in small gaps the power transfer via heat radiation is calculated directly by the program, the calculations of the heat radiation in larger hollow spaces and in the pellet feeder of the core, as well as the calculations of the natural convection processes in hollow spaces are carried out by means of the effective heat conducting capacity. The power is determined on the basis of the heat source density in the meshes of the relevant material zones. By means of relevant sub-programs, the program user can also use the temporal and place benefits of the heat source density distribution in a more flexible way. Taking the start-up temperature distribution in account is also possible.

In our opinion the TAC2D program is suitable for analyzing the temperature development in the case of a core warm-up after DES, since the relevant physical and geometric effects are recorded. This fact is also confirmed by a verification performed by the applicant in the form of a comparative calculation between the TAC20 and THERMIX programs with the same model inputs.

THERMIX is generally considered to be the best suitable available program for analyses of thermodynamic behavior of pellet heap reactors (L 84). This results from the investigations performed by KPA-Jülich in the case of a pressure relief breakdown. The investigations mainly dealt with the thermodynamic behavior of the HTR module (L 128, L 129, L 141). All these investigations were performed using the THERMIX program. According to the selection of boundary conditions, the investigations provided approximately the same results. This valid especially for the temperature processes in the core and in the structures around it (reflectors, the core container, the reactor pressure vessel).

Taking into account the fact that the applicant has checked temperature development under depressurised conditions, the most important value to be checked was the maximum core temperature. The result of this comparison shows good compliance with the calculated maximum core temperatures /U 5.4-2/ to /U 5.4-5/. Those significant differences in temperatures in the area of the spherical pellet edge are in our opinion of inferior technical-security importance.

Analysis

The temperature calculation with respect to the DES was provided by the applicant with the help of the modified TAC2D-version /U 5.4-6/. This program version contains, in comparison with the standard version, a simplified one-dimensional unsteady flow component for the calculation of the initial temperature distribution. At the same time, the transport of energy between the solid material and the liquid was determined to be similar to the calculation program THERMIX /L 104/, that is, the enthalpy taken from the liquid is calculated on the basis of the difference between the surface temperature of the solid material and the temperature of the liquid, as well as of the local thermal conductivity coefficient.

Inside the solid material, the enthalpy is absorbed or emitted by the liquid is taken into consideration as heat drain or heat source. A eventual effective nuclear heat source is superposed. In contrast with THERMIX, the energy transmission in the liquid is provided only in the axial direction.

The solid material is considered homogenous, whereas the material parameters can be corrected to comply with the porosity. The local material data, for example, the thermal conductivity and the thermal capacity will be considered among others as dependent upon temperature and, if need be, upon quantity.

The thermal transmission in the pellet heap takes place via the following mechanisms:

- the thermal conductivity in the pellets and via contact places to the adjoining pellets,
- the thermal conductivity via the gas in the gaps between the pellets,
- the thermal radiation between the adjoining pellets

These thermal transmission mechanisms are integrally approximated as a net thermal conductivity event with an effective thermal conductivity as a proportional constant. At the same time, the values for the effective thermal conductivity of the depressurized non-flowed through pellet filling are determined for temperatures below 1,300°C in accordance with Zehner-Schlünder /L105/ and above 1, 300°C in accordance with Robolt /L140/. They are dependent on geometrical data, the material parameters of the liquid, as well as the radiation emission grade, the thermal conductivity, and the temperature of the pellets. A constant value of 17W/mK is used for the thermal conductivity of the pellets in accordance with /U 5.4-1/.

The power capacity to be discharged depends on the axial and radial power distribution before the breakdown occurrence and the time slope of the additional thermal power, which is indicated for the different axial core positions. With the help of an interpolation in time and position results then the power in any core mesh.

The facility model on which the temperature calculations are based consists of 44 radial and 72 axial meshes, of which 11 radial meshes and 26 meshes in the axial direction are eliminated. The filling pellets on the free surface and on the floor reflector are not shown.

Considering the unfavorable starting conditions and the error influences, the applicant calculates a maximum temperature of the fuel elements of 1,608°C, which is reached within approx. 30 hours after the breakdown occurrence. Based on the thermal resistance and the accumulation and removal conditions of the core and the adjoining structures (reflector, coal-stone, hollow spaces), the heating of the side reflector, the core container, and the RDB is delayed. Therefore, the side reflector reaches a maximum temperature of 860°C /U 5.4-1/ after approximately 75 hours and the core container reaches a maximum temperature of 490°C /U1, U 5.4-6/ after approximately 100 hours. The maximum temperature in the central RDB-section is after about 90 hours approximately 350°C /U1, U5.4-6/ and in the lower RDB-section, due to the insulation provided, it is after about 150 hours approximately 355°C /U 2.6.2-8/. Because of the redistribution of the axial temperature profile during the heating of the core, the floor reflector is cooled for approx. 300 K, whereas the temperature of the ceiling reflector rises to values under 500°C within approx. 90 hours.

The effectiveness, and therefore the temperature of the area cooler, does not have, in accordance with /U 5.2-1/, any significant influence on the maximum fuel element temperature in the case of a core heating failure. This is confirmed by our analysis as well as by the investigations made by KFA-Jülich /L128,L 129/.

Because of special importance of the DES with regard to the temperature development in the core and in the equipment of the RDB, we have carried out our own analyses. The standard version of the program TAC2D /L139/, which we used, has been primarily modified by us by expanding the field magnitudes, so that additional meshes can be provided in comparison with the standard version.

The plant model, which we used, is shown in Figure 5.4-1. It is rotation-symmetric (R-Z-Geometry) and consists of 54 grid lines in the radial direction and forms the area from the core axis (R=0m) to the external side of the primary cell (R=6,975m). In the axial direction, the model consists of 96 grid lines and comprises the area from the upper concrete layer of the cell primary cover (Z=-9,969m) to the lower concrete layer of the cell primary floor (Z=19,50m), whereas the upper edge of the core corresponds to the reference point (Z=0m). Due to program–technical reasons, the Z-axis is shown shifted by + 10.669m in Figure 5.4-1, because the program cannot process any negative coordinates. The reactor core itself consists of 12 radial and 23 axial grid lines.

For the whole plant model therefore result 5,335 internal and external local supporting points, for which the temperatures in every time step must be determined via the release of the thermal conductivity comparison and, if need be, the material values and power values must be confirmed. The plant model consists of 68 different blocks altogether (including four blocks for the edging) and these again consist of 12 different materials, of which the material parameters will be calculated by different subordinate calculation programs. These sub-programs for the calculation of the material parameters and for the power distribution and development were tested in the respective parameter range with separated main programs in order to identify and eliminate program errors.

The results of the temperature calculations depend mostly on the assumptions of the respective input variables. Due to the great importance of these input variables for the analysis results we have summarized them and noted at the same time in which cases we have reached different assumptions than the applicant.

It is necessary to consider the influence of the systematic and statistic errors of the input variables in accordance with the requirements of the KTA-regulation 3102.5 /L 48/. This takes place via a parameter variation in the temperature calculations. The following unfavorable conditions have been considered for reference.

- As initial power was established as reference a thermal reactor power of 210 MW corresponding to 5% of overload. This overload is based on the assumed reactor power limitation.
- The initial temperature distribution is based on an increased cool gas temperature of 280°C and an increased hot gas temperature of 750°C in accordance with the corresponding RES limitation values. /U 5.4-8/.
- In order to determinate the remaining available power emission of delayed neutrons during the breakdown initial phase immediately after the RESA, we took into consideration a power integral of three full-load /VLS/ seconds.

With these assumptions for this reference case, a maximum core temperature of 1,535°C results from our calculation.

In addition to the unfavorable already assumed initial condition in the reference case. the following influences were considered as systematic errors:

- The unfavorable assumption that the reactor high-speed shut-off system is triggered only from the second initial impulse "negative sliding limit value of the thermally corrected neutron flow ≥ 20%/min."

As per the applicant's investigation /U 5.4-1/, the RESA triggers only via the second impulse, an increase of the accumulated energy of less than 4.5 VLS. Therefore, in contrast with the reference case, we took into consideration an increase of the accumulated energy by 5 VLS and determined the rise of the maximum core temperature of 2 K therefor.

- Production-related small gas gaps can occur in the ceramic RDB built-in components /U 2.5.3-2/. These gas gaps are not considered in the reference case. The assumption of a longitudinal vertical gap of 1mm between the side reflector and the coal stone results in a temperature rise of 2 K.

The determination of the influence of the statistic errors was based primarily on 1δ deviation of the individual influence variables in order to be able to proceed with the statistic fluctuation for the expectation value of normal and equalized influence variables in the same way. At the same time, in the case of an equally divided random variable with a fluctuation range of ± h whose standard deviation is determined as the square root of three of the central moment of second order /L 142/ to $h/\sqrt{3}$. The influence of the following entry sizes will be considered as statistical errors:

- The inaccuracy in determining the time curve of the after-heating phase is based on the consideration of an error of 5.6% corresponding to a 2 δ deviation (see Chapter 2.5.5). In the case of a 1 δ deviation corresponding to 2.8% results an increase of the maximum core temperature of 21 K.

- In the stationary local power redistribution is based on a power excess of 5% in the sense of a 1 δ deviation (see Chapter 2.5.5). The power excess extends approximately ± 70cm beyond the place of the maximum core temperature and causes a temperature increase of 25 K.

- As a nominal value for the thermal conductivity of the fuel elements is used as basis the value of 17W/mK in accordance with /U 5.4-9, U 5.4-10/.(see Chapter 2.5.5). It is obvious from the measurements that the measurement values are equally distributed with a fluctuation range of 2 W/mK for this expectation value /U 5.4-1/. An increase of the maximum core temperature by 27 K, which corresponds to 16 K for a 1 ð deviation, emerges from this decrease of the thermal conductivity.

- The effective thermal conductivity of the non-flowed through pellet filling is temperature-dependent and will be calculated with the help of a polygon. In accordance with Robold /L 140/, a 1 ð the deviation in the respective temperature range is equal to an error of 5%. The decrease of the effective thermal conductivity by 5% causes an increase of the maximum fuel element temperature of 22 K.

- The data of the temperature-dependent thermal accumulation of the pellet filling deviate with an equipartition of ± 5% /U 5.4-1/. The reduction of the thermal capacity by 5% causes a temperature increase of 11K, if need be, of 6 K, with a 1 ð deviation.

- In the case of a thermal conductivity of the reflector-graphite, a reduction of 5% considered by us causes a core temperature increase of 2 K, which we consider as a 1 ð deviation.

- Assuming that the deviations of the thermal conductivity of the reflector material are also equal to an equipartition with a dispersion width of ± 5% results a standard deviation of 2.9%. A reduction of the thermal capacity for this deviation causes a core temperature increase of 1 K.

- The effective thermal conductivity in the gas-filled hollow spaces consists of a radiation and convection factor. In particular the convection factor is based on semi-empirical correlation's /L 143/ for which we assume an error of 30 %. Considering a reduction of 30% for the convection factor and 10% for the radiation factor in all the gas-filled hollow spaces results an increase of the maximum core temperature of less than 1K.

- The measuring devices for the cool and hot gas temperature have a standard deviation of 0.5% with respect to 1.2 times the values measured in Celsius degrees /U 5.4-1/. In this way results a measurement error of approx. 2 K for the cool gas temperature and of approx. 5 K for the hot gas temperature. An increase of the initial temperature distribution (250 C/700 C) to the RESA response values (280 C/750 C) causes an increase of the maximum core temperature of 19 K. Taking into considering this temperature increase, we estimate the influence of the aforesaid measurement errors on the maximum core temperature increase to be of 2 K.

Other error influences caused by further outwardly installed structures, for example, inaccuracies of the material data of the coal-stone, the core container, and the RDB, or the temperature of the surface cooler, which was considered in the analyses with a temperature of 40 °C, are not measured in the error analysis.

These influences do not show any significant effects on the maximum core temperature as proved by the respective investigations, which were carried out by the applicant and by us.

When statistically superposing the individual error contributions, our analysis shows, when considering a 1 ð deviation, a statistical total error of

T1 ð= square root of $\sqrt{21^2 +25^2 +16^2 +22^2 +6^2 + 2^2 +1^2 +1^2 +2^2} = \pm 43$ K

if need be, considering a 2 ð deviation, of

ΔT 2 ð= ±86 K.

Based on the core temperature for the reference case and superposing the systematic and statistical error contributions, the maximum core temperature is

T_{MAX} = 1 535 °C + 4K + 86K= 1 625°C.

Table 5.4-1 summarizes the significant influence sizes for the core temperature, their 1-ð deviations, and their effects on the maximum core temperature.

Moreover, the most important results of our temperature calculations for the reference case are shown in Figures 5.4-2 to 5.4-12. For different time points, the maximum temperature curves are shown on the core axis (R=3,25cm) in Figure 5.4-2, for the position of the reflector rods (R=157,5cm) in Figure 5.4-3, in the core container wall (R=272cm) in Figure 5.4-4, and in the RDB wall (R=297cm) in Figure 5.4-5. In addition, the radial curves of the temperature in the core center at different time points (Z=522,05cm) are shown in Figure 5.4-6, at the height of the maximum RDB temperature (insulation, Z=1043,1cm) are shown in Figure 5.4-7, in the bottom structure of the core container (Z=1373,1cm) are shown in Figure 5.4-8, and in the RDB bottom (Z=1605,6cm) in Figure 5.4-9, wherein the core upper edge corresponds to the reference point (Z=0cm). The temperature curves related to time for different core positions are shown in Figure 5.4-10, at different locations of the RDB wall in Figure 5.4-11, and for different reflector and core container positions in Figure 5.4-12.

The highest local core temperature for the reference case was determined by the applicant as 1,522°C and by us as 1,532°C (see Figure 5.4-10). Taking into account the explained error contributions, a maximum core temperature of 1,608°C was confirmed by the applicant and of 1,625°C by us. The error analysis considered by the applicant in this connection contains the significant influence variables and thus meets the requirements of the KTA Regulation 3102.5. To cover everything, we should mention the fact that an error that the numerical solution method has was not considered. In our opinion, this is acceptable in this case because, in accordance with the applicant's investigation /U 5.4-1/, when the time and local fluctuations are varied, it results that an error of the numerical solution method deviates systematically upwards, that is, the calculated values overestimate the factual result.

A comparison of our calculations and the applicant's calculations of the temperature development in the core shows that the results are in agreement both from the quantitative and qualitative point of view. The deviations of less than 20 K in the case of the maximum core temperature are most likely caused, in our opinion, by the different plant models and the differences in the investigation of the material parameters. In addition, the applicant has proceeded on the assumption of a maximum fuel element temperature of 1,620°C, which is confirmed by us as a temperature 1,625°C, wherein we consider the deviation of 5 K in this connection to be unimportant.

We have also taken into consideration the fact that, according to our reference case, with a maximum core temperature of 1,535 °C, only a smaller portion of approx. 1% of the core volume reaches a temperature of more than 1,500°C, and a portion of approx. 7% reaches a temperature of more than 1,400°C.

However, we mention the fact that the temperature calculations provided by us and by the applicant establish an initial condition which is established in particular by the base power distribution of 105% /U 2.5.5-1/ and the base temperature distribution /U 5.4-8/ while considering the RESA limit values for cool gas temperature (280°C) and hot gas temperature (750°C). Therefore, it is necessary to observe and adhere to the established assumptions with regard to the reactor limitation and the temperature distribution in the reactor core during a further plant and equipment planning. In this connection, the RESA limit values, the influence of the bypasses, the temperature strands, and the cooling up to the measuring point as well as a calibration and a 2-ǒ-measurement error of 1% of the whole measurement chain is also considered. In this connection, we draw attention to the document /U 5.4-7/ in which the influence of the bypasses on the temperature distribution in the core as well as the concept of operational determination of these bypasses and their consideration are shown.

The temperatures in the metal built-in components of the RDB are shown for reference in Figures 5.4-2 to 5.4-12. In addition, we have summarized for this case the maximum temperatures for these built-in components determined by us in Table 5.4-2. A comparison with the temperatures determined by the applicant shows that the temperatures calculated by us are approx. 10 K to 20 K higher and therefore correspond to the trend of core temperatures. Only in the lower RDB, in the area of the provided insulation (see Figures 5.4-5, -7, -11), we have calculated an RDB temperature of 402°C, which is significantly higher than the temperature determined by the applicant 355°C /U 2.6.2-8/. Aside from the already mentioned differences of the plant model, a substantial reason for these deviations would be the different way in which the thermal conductivity of the insulation are taken into consideration. Because in this case a constant value of 0,1W/mK is assumed /U 2.6.2-8/, we have applied a temperature-dependent formula /U 4.5.4-11/, which is the reason for the lower values in the respective temperature range.

We do not consider the substantial excess of the design temperature in the insulated RDB area as conceptually decisive because the temperature distribution in this area is not so dependent on the heat transmission from the core, but rather on the type and geometry of the insulation. However, we must mention the fact that in further plant planning, the type and geometry of the insulation must be optimized, so that the permissible RDB temperature, even taking into account the inaccuracies of the input variables, will not be exceeded. For other information related to this, see Chapter 2.6.2, wherein an evaluation of the strength properties of RDB is undertaken.

In the area of the reflector rod, the maximum temperature we have calculated is 878°C (see Figures 5.4-3,-6,-12), which exceeds the temperature of 860°C established by the applicant /U 5.4-6/. Since both temperatures are substantially higher than the design temperature of 650°C, we do not consider the deviation of 20 K between the maximum temperatures as substantial.

In this case, please refer also to Chapter 2.5.5.2, in which the thermal design of the reflector rods is described.

The maximum temperatures of the core container have been determined by the applicant as 490°C /U1, U5.4-6/ and by us as 501°C (see also Figures 5.4-4,-6,-12), so that the design temperature of 500°C of the core container can be considered as maintained

The maximum hot gas temperature of this breakdown corresponds to the initial temperature and is approximately 750°C: it is substantially below 900°C, which is the design temperature of the gas tubes of the hot gas pipeline.

The loads on the reactor body caused by pressure and temperature increases are reviewed in Chapter 5.4.4.

The applicant's investigation of the thermo-hydraulic characteristics of the reactor has revealed that the breakdown "a rupture of a large connecting pipe" results in the highest temperatures in the reactor core and the adjoining structures and it can be considered that it covers all the smaller leakage and rupture events. This is in agreement with our results and the investigations of KFA-Julich /L 128, L129, L141/.

Table 5.4-1: Systematic and statistical influence of errors on the determination of the maximum core temperature

Including the reference case	1,535°C
- Initial power 105%	
- RESA GW for cool and hot gas temperature 280°C/750°C	
- 3 seconds at full load	
Systematic influences	
- 5 seconds at full load (2nd RESA triggering)	2 K
- Gap between the reflector and the coal-stone (1mm)	2 K
Systematic total error ΔT_{sys}	4 K
Statistic influences (1-ð deviations)	
• Time curve of the after-decay power (1 ð deviation 2,8 %)	21 K
• Local power distribution (+ 5%)	25 K
• Thermal conductivity of fuel element (-1,15W/mK)	16 K
• Effective thermal conductivity of core (-5%)	22 K
• Thermal accumulation of core (-2,9%)	6 K
• Thermal conductivity of reflector (-5%)	2 K
• Thermal accumulation of reflector (-2,9%)	1 K
• Effective thermal conductivity in hollow spaces (-30% for convection and -10% for radiation)	1 K
• Measurement error of gas temperatures (+0,5%)	2 K

Statistic total error

$\Delta T_{1\delta}$= square root of $\sqrt{21^2 + 25^2 + 16^2 + 22^2 + 6^2 + 2^2 + 1^2 + 1^2 + 2^2} = \pm 43$ K

$\Delta T_{2\delta} = \pm 86$ K.

maximum core temperature

$T_{max} = 1{,}535\ °C + 4K + 86K = 1{,}625°C$

Table 5.4-2: Maximum temperatures in the metal RDB built-in components in accordance with DES

Reflector rods (Z=522.05cm/R=157.5cm)	878°C (70 h)
Core container - wall (Z=522.05cm/R=272cm)	501°C (85 h)
Core container - bottom (Z=1,373.1cm/R=35cm)	378°C (240 h)
RDB- centre (Z=522.05cm/R=297cm)	359°C (90 h)
RDB-insulation (Z=1,043.1cm/R=297cm)	402°C (140 h)
RDB-bottom floor (Z=1,605.6cm/R=190cm)	350°C (300 h)

Picture 5.4-1: Plant model -HTR module

Vertical axis: TEMPERATURE (°C)
Horizontal axis: CORE AXIS (cm)

STD=HOURS

AXIAL TEMPERATURE CURVE WITH R = 02 (3.25cm),
FIGURE 5.4-2 CORE HEATING ACCORDING TO DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: CORE AXIS (cm)

STD=HOURS

AXIAL TEMPERATURE CURVE WITH R =15 (157.5cm),
FIGURE 5.4-3 CORE HEATING IN ACCORDANCE WITH DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: CORE AXIS (cm)

STD=HOURS

AXIAL TEMPERATURE CURVE WITH R=30 (272.0cm),
FIGURE 5.4-4 CORE HEATING IN ACCORDANCE WITH DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: CORE AXIS (cm)

STD=HOURS

AXIAL TEMPERATURE CURVE WITH R =35 (297.0cm),
FIGURE 5.4-5 CORE HEATING IN ACCORDANC WITH DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: RADIUS (cm)

STD=HOURS

RADIAL TEMPEREATURE CURVE WITH Z=51 (522.05cm),
FIGURE 5.4-6 CORE HEATING IN ACCORDANCE WITH DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: RADIUS (cm)

STD=HOURS

RADIAL TEMPERATURE CURVE WITH Z=65 (1043.1cm),
FIGURE 5.4-7 CORE HEATING IN ACCORDANCE WITH DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: RADIUS (cm)

STD=HOURS

RADIAL TEMPERATURE CURVE WITH Z=78 (1373.1cm),
FIGURE 5.4-8 CORE HEATING IN ACCORDANCE WITH DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: RADIUS (cm)

STD=HOURS

RADIAL TEMPERATURE CURVE WITH Z=85 (1605.6cm),
FIGURE 5.4-9 CORE HEATING IN ACCORDANCE WITH DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: TIME (HOURS)

TEMPERATURE CURVE IN THE CORE
FIGURE 5.4-10 CORE HEATING IN ACCORDANCE WITH DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: TIME (HOURS)

TEMPERATURE CURVE IN THE CORE
FIGURE 5.4-11 CORE HEATING IN ACCORDANCE WITH DES, RUN 37

Vertical axis: TEMPERATURE (°C)
Horizontal axis: TIME (HOURS)

TEMPERATURE CURVE IN THE CONTROL RODS AND CORE CONTAINER
FIGURE 5.4-12 CORE HEATING IN ACCORDANCE WITH DES, RUN 37

5.4.1.2 Break of a Small Pipeline and Small Leaks

In the security report are also considered the cases of

- break of a small metering line,
- small leaks,
- improper opening of a safety valve.

The small metering lines have an inner diameter of less than 10 mm. The break of such a line is recognized by the reactor safety system and a RESA is triggered. The prescribed shut-off valves outside of the primary cells are not automatically closed in such a case by the reactor safety system so that one can at all times obtain information about the condition of the reactor. The escaping primary cooling agent is detected in the exhaust air and filtered through the secured pressure system and the furnace and released into the environment. In the case of leaks behind the shut-off valves, a leak shut-off can be executed manually. When unable to shut-off the leaks, a relatively slow reduction in pressure occurs up to the environmental pressure. During this phase of pressure reduction, the plant can be shut-off through the secondary side.

Even small leaks in lines carrying primary cooling agents can initiate such a minor pressure reduction that a recognition thereof by the reactor safety system is at first not possible, since a saturation takes place via the pressure regulation. In these cases, the escaping primary cooling agent is also detected by the ambient air control, which leads to notification to the control room. The reactor shutdown and possibly other safety measures can then be initiated manually based on the time available.

The two primary safety valves (SiV) open at a recognized pressure of 69 bar (1st SiV), that is, of 72.5 bar (2nd SiV), wherein the opening diameter corresponds to an interior diameter of DN 10 (1st SiV), that is, DN 65 (2nd SiV). An activation of the 1st SiV occurs in the failures of a steam generator heating pipe break with an additional malfunction of the operational systems (see Chap. 5.4.3) and long-term back-up generator drop (see Chap. 5.3.4). The safety valves close themselves automatically again after reaching a minimum pressure. In the case of a blockage of the opening or a leakage as a result of an incomplete closure, a pressure containment valve aligned with the respective safety valve closes as soon as the primary circuit pressure of 8 % lies below the operational pressure.

In the case of a secondary failure of the valves, a slow but also a faster pressure relief, depending on the exception constellation of the primary circuit design, can occur and enter into the reactor building through the ventilation (see Chap. 2.6)

In the case of small leaks in the lines carrying primary cooling agent, it is to be differentiated between leaks, which are promptly recognized by the reactor safety system, and those, which are not or only later recognized. The breaks of small leaks still recognized by the reactor safety system occur more slowly than in the case of a break of a large connecting line. But, in the end, the same countermeasures such as the RESA and the primary core shutdown are triggered under the same peripheral conditions. Because of the slower pressure release in the primary system and the consequently occurring better convective heat exhaust, the temperature load remains below that of a break of a large connecting line.

The even smaller leaks, which are not or at least not immediately recognized by the reactor safety system, are detected by the ambient air monitoring as an activity increase and are reported via signals to the control room. Afterward, countermeasures such as the leak blockage, the RESA, and the shutdown via the secondary side can be initiated manually. Because of the very slow process of the failure decline, there is sufficient time for these countermeasures and consequently we view the control of the failures via manual measures as reliable.

The illustration of the failures as a consequence of small leaks, breaks of small connecting lines, and improperly opened safety valves was considered qualitatively in the safety report. Even though we view such an illustration of the conceptual evaluation as sufficient, we point out that within the context of the further planning of the design of the reactor safety systems, quantitative security measures should be undertaken. At this time it is to be determined up to which size of leaks the automatic reactor safety measures are to be initiated even when considering a single failure and the absence of the 1st RESA-stimulation.

In summary, we find that we do not assign a conceptually decisive importance to the small leaks discussed in this chapter, since they are of subordinate importance with a view to the cooling of the core. We examined and evaluated the emission of radioactive materials into the environment via the failures caused by small leaks in Chap. 5.8.1.

5.4.1.3 Breaks or Leakage of Lines Conducting Primary Cooling Agent Outside of the Reactor Building

In the case of breaks in pipelines carrying primary cooling agents outside of the reactor building, the primary circuit shut-off valves are triggered by the reactor safety system and thus the primary circuit is separated from the leak. In this respect, the break in the line with the greatest nominal width (DN 65) such as, for example, the break of a pipe of the helium purification plant, should be seen as the determining factor. In this instance, the primary cooling agent from the defective purification strand flows into the building and until the check valve is closed also from the primary circuit in the building. This leads to the activation of the pressure relief valve in the reactor ancillary building and thus to the release of unfiltered primary cooling agent. Afterward, the relief valve closes again, and the release occurs via secure pressure conditions and the chimney.

In the case of the secondary failure of the primary circuit check valve as a result of a single mistake or in the absence of the 1st RESA stimulation, it comes to a rapid pressure release of the primary circuit via the negative sliding limit value of the primary circuit. The uncontrollable break of a large connection pipe and the effects on the reactor core and the RDB built-in components are covered in Chap. 5.4.1.1. The emission of radioactive materials into the environment is covered separately in Chap. 5.8.1.

5.4.1.4 Summary Evaluation of the Primary Side Breaks and Leaks

The uncontrollable double-end break of the largest pipeline of primary cooling agent with a nominal width of DN 65 directly at the pressure vessel unit leads to complete loss of primary cooling agent. This failure case applies for all the other subordinated breaks and leaks.

In an independent comparison calculation, taking into consideration the unfavorable peripheral conditions, we examined the adherence to the design temperature of 1,620 ° C of the fuel elements. We view the minimal excess of this temperature as determined by us as unimportant, whereas we also consider that only a minimal number of the fuel elements reach temperatures in the area of the design temperature.

According to our calculations, the design temperatures of a few components are minimally exceeded. In our opinion, this is not relevant with respect to the concept and can be taken under consideration in the area of construction planning.

As a result of the loss of cooling agent and the following core temperature increase, radioactive materials can be released. The resulting radiation exposure in the environment remains, even in the absence of the anticipated filtering, far below the planned standard value of § 28 of the Radiation Protection Regulation.

We consider that the anticipated concept is appropriate to control breakdowns as a result of related primary side breaks and leaks.

5.4.2 Secondary Side Breaks and Leaks

Paragraph 5.1 discloses the triggering events, which initiate the secondary side leaks and breaks.

These include:

- break of the feed water line,
- break of the fresh steam line

A break of the heating pipe is evaluated in Chapter 5.4.3.

The secondary side pipelines – the feed water line and the fresh steam line – are connected outside of the steam generator cell to the feed water line or the fresh steam nozzle of the steam generator. Both nozzles are components of the pressure vessel unit on which for reasons of quality assurance measures no malfunction is presumed, so that breaks at the feed water supply and fresh steam lines should be investigated. A simultaneous occurrence of damage in the heating pipe of the steam generator is not expected.

The feed water and fresh steam lines can be arranged in sets of two in a housing block, connected in series, self-activating quick closing valves, which are controlled by the reactor safety system. The closing times are approximately 2 seconds. In addition, it is possible to block feed water side via a check valve behind the feed water check valve.

Break of the Feed Water Line

A 2F break of the feed water line is recognized by the reactor safety system by the following stimulus criteria:

- the throughput ratio (primary and secondary side),
- the negative sliding limit value of the fresh steam pressure greater than/equal to 8 bar/ min.

The break leads to an abrupt interruption of the feed water throughput and to a flow reversal of the steam generator. The interruption of the feed water throughput causes the immediate response of the 1st criterion of the throughput ratio (primary to secondary side) greater than/equal to .3. In the absence of this criterion, the shutdown takes place a short time later due to the exceeding of the threshold limit for the fresh steam pressure. With the shutdown of the reactor (descent of the reflector rods, shutdown of the blower), the takes place the shutoff of the secondary circuit.

When there is a break in a feed water line connected to the feed water nozzles, can be differentiated the following cases in accordance with the location of the occurrence:

- a break before the feed water check valves,
- a break behind the feed water check valves (between the feed water valves and the steam generator).

When there is a break before check valves, the steam generator is blocked after the flow reversal via the check valve, that is, within 2 seconds, via the feed water check valves. On the other side of the break takes place a long-term outflow of feed water from the feed water container into the reactor building.

A break behind the feed water check valves cannot be shut off on the side of the steam generator and leads to the emptying of the steam generator into the reactor building. The break side connected to the feed water container is closed via the check valves.

Break of the Fresh Steam Line

A 2F break of the fresh steam line (FDL) is recognized by the reactor safety system via the stimulus criteria

- negative sliding limit value of the fresh steam pressure greater than/equal to 8 bar/ min.

The break leads to a steep pressure drop in the steam generator, thus the reactor safety signal already activates within seconds and triggers the actions "descent of the reactor rods," "shut down of the blower," and "shutoff of the secondary circuit." A second recognition criterion of the reactor safety system for the FSL break is not available. However, the applicant later affirmed in a supplementary document /U 2.9-6/ that this signal can be assigned a higher priority, so that a failure of the signal cannot be precluded.

Also in the FDL break should be basically taken into consideration the two break locations:

- a break before the FD check valve (between the steam generator and the FD valves),
- a break behind the FD check valves.

A break of the FDL before the check valves cannot be shut off on the side of the steam generator, so that a complete emptying of the steam generator follows. Via the shutoff triggered by the blockage of the feed of the steam generator is prevented a long-term outflow.

The outflow out of the break end connected to the second module via the fresh steam line is shutoff via the simultaneously triggered closing of the fresh steam check valves.

When there is a break in the fresh steam line behind the FD check valves, no outflow into the reactor building takes place, since the FDL is configured between the valve block and the reactor building wall lead-through as a double pipe construction /U 2.9-5/.

In his specifications for the FDL break, the applicant merely starts with a break condition before the check valve. In this way results an outflow out of the blocked steam generator as well as up to the closing of the FD valves [and] an outflow out of the break end, which is connected to the second module.

After the failures "break of the feed water line" or "break of the fresh steam line," the primary circuit is under pressure. The secondary side heat exhaust is interrupted. The heat discharge occurs via the flat surface coolers. In the long run, the process of these failures can be compared with a manual RESA. The design limits are not reached. The effects of these failures on the primary circuit are comparable to the corresponding effects in the RESA.

In the short run, a break in the feed water line equals an immediate interruption of the heat exhaust. Thus, the primary circuit temperatures increase minimally up to the time of the RESA.

This increase, however, has a neglectful influence on the maximum temperatures, which in the long run occur after the RESA.

When there is a break in the fresh steam line can be expected a short-term decrease of the hot gas temperature as a result of the increased steam exhaust. Since, however, the afterflow of the feed water is limited via the throttles in the steam generator and a blockage of the feed water side follows, the temperature drop is minimal. The possible retroactive reactions are lower than in discussed case of the "decrease of the cold gas temperature."

We consider that the initiation of reactor safety actions after a break in the fresh steam line merely via a stimulus criterion as permissible, when this stimulus criterion is executed on a higher priority in the sense of the KTA Regulation 3501.

Chap. 5.4.4 addresses the occurring loads on the building in the case of a break of the lines mentioned above.

5.4.3 Damage to the Steam Generator

The penetration of water into the core should be seen as a design flaw in the case of high temperature reactors.

In this case, because of the higher pressures of the secondary side, water in liquid or gaseous form, penetrates into the primary side of the steam generator. If this water reaches the reactor pressure vessel it comes to an increase of the reactivity and corrosion of the graphite built-in components and fuel elements. In this way is generated water vapor, a mixture of H_2 and CO.

In the following are evaluated the amount of water to be considered and the resulting corrosion. The increase of reactivity is discussed in Chap. 5.2.5.

A special investigation of the emergency power failure with breaks in the steam generator heating pipe is not required, since operational systems which would have to be supplied with emergency power in this case, are not needed. The case of the emergency power failure does not aggravate a water penetration into the primary circuit. The influence of the emergency power failure on the maximum core and component temperatures is described in Chap. 5.3.4.

The following evaluation criteria were applied to the superceding regulations and directions mentioned in Chap. 5.1:

- For each breakdown of the reactor safety system to be controlled, at least two physically different response criteria have to be called upon. If the requirement cannot be reasonably or technically met, different measurement processes for the measurement recordation, different measuring devices in the respective response channel groups, as well as shortened test cycles or equal measures must be provided.

- The reactor core and the important technical safety-related plant parts and systems must be designed in such a way that, while considering the maximum water entry into the primary circuit, the specified limits values of the loads of remain within the allowed limit values. This is the case when the fuel element temperatures remain limited to values of approximately 1,620 °C and the loads on the technically safety-relevant plant parts and systems remain below the values of the design limits.

- The graphite corrosion must be limited in such a way that the water gas concentration lies below the ignition limit.

- The exposure of radiation caused by the emission of radioactive materials from the primary circuit into the environment must remain below permissible values.

Analysis Methods

The applicant uses the program GRECO /U 5.4-17/ for determining the graphite corrosion.

For determining time curve of the graphite corrosion and the formation of water gas after water a penetration of water, the circuit is divided into single volume elements, in which the masses and partial pressures of the participating gases (He, H_2O, H_2, etc) and the temperatures of the gas mixtures are defined.

The timely variable temperatures of the core and the reflectors, the temperature curves in the circuit sections, and the throughput in the primary circuit are a predetermined as boundary conditions.

As sources for the individual gas types are taken into consideration the graphite corrosion (H_2,CO, CO_2) and the pressure regulation (He), as drains are taken into consideration the corrosion (H_2O), the water separator, the gas purification, the pressure regulation, and the exhaust ventilation via safety valve.

As a result, the program delivers the time curves of the partial pressures or masses of the participating gases in the circuit sections, the corrosion rates in the core and the outflow rates via the safety valve. Additionally, the composition of the escaping gas mixture is given for evaluation of ignition capability.

The applicant calculates the penetration of water into the primary circuit with the program PSS III, under the following assumptions /U 5.4-12/:

- a heating pipe break under full load operation
- a blockage of the steam generator within 1 sec. after stimulus, delay of fresh steam related blockage by 1 sec.,
- steam generator relief only via one strand,
- an opening of the relief valve (NW 50) within 3 sec.,
- a closing of the relief valve at when there is pressure balance with the primary circuit

According to this assumption, in a double-ended break of a steam generator heating pipe, water escapes at 1.4 kg/s and steam at 3.9 kg/sec. The water which penetrates into the primary circuit is detected within 10 seconds according to the criterion “moisture in the primary circuit approx. ≥ 800 vpm.” The safety actions are triggered by the reactor safety system (descent of the reflector rods, shutdown of the primary circuit ventilation, blockage of the secondary circuit, and steam generator discharge).

Up to the blockage of the steam generator, the flow of the break mass remains constant, since the small flow of break mass can be easily be refilled afterward in comparison with the secondary throughput. After a blockage of the steam generator and an opening of the relief valves, the pressure in the steam generator and in the flow of the break mass drops continually.

According to the calculation results disclosed in the supplement /U 5.4-12/, the pressure compensation of 60 bar in 37 sec. is achieved after the stimulation of the reactor safety activities. Up to this point and until the stimulus of the reactor safety system, 54 kg of H_2O, and additionally 117 kg of H_2O were transferred during the steam generator relief to the primary circuit. After achieving pressure compensation, the steam generator is separated within seconds by closing the relief valve. A remnant of 300 kg of H_2O remains in the steam generator. It is assumed that this remnant will be completely released into the primary circuit. Thus, according to the calculations of the applicant, a total of 471 kg of water passes to the primary circuit. For a further examination of the effects of a break of the steam generator heating pipe, the applicant submits that a total of 600 kg of water pass to the primary circuit.

The penetrating water reacts with the graphite and leads to corrosion of the fuel elements and the ceramic built-in components.

The breakdown of the water separator of the helium purification plant is shut off manually. The water gas (H_2 and CO) and the residual moisture, which were generated, are removed from the primary circuit (3 nag/h). A demand on the pressure relief system on the primary side is thus prevented.

In the absence of the possibility of shutting down the failure of the water separator of the pressure regulation and the helium purification plant and with the continuously open blower valve, the pressure increases in the primary circuit. At 69 bar, the safety valve of the pressure relief system opens after approx. 3.5 hours and for approx. 30 minutes exhausts a mixture of primary cooling agent, water gas, and steam into the reactor building. After 4 hours, the converted graphite mass amounts to 300 Kg. The mixture flowing into the reactor building amounts to 10 % of the primary circuit volume. The water gas fraction is around 9 %. The maximum corrosion rate of the fuel elements amounts locally to approx. 3.5 % /U 5.4-13/.

We checked the analysis of the applicant with our own calculations, with the object of determining the mass of water penetrated into the primary circuit.

The greatest outflow rate in the break of a double-ended heating pipe occurs with a break directly before the fresh stream pipe plate. The outflow rate of the feed water is limited via the stabilizing throttle at the inlet of the heat pipe. By using the loss of pressure specified by the applicant via the throttle at 10 bar /U 5.4-12/, we were able to confirm a stationary outflow rate of 1.4 kg/sec of water. The effects of the throttles on the directional flow in the secondary circuit (degree of effectiveness) are not the object of this assessment.

For the stationary outflow rate of fresh steam by 3.75 Kg/sec, we calculated an insignificant lower value than the applicant did. In this way, the total mentioned stationary outflow rate of 5.3 Kg/sec can be verified. This value applies to the full-load and partial-load operation.

During the start-up and shutdown, according to the start-up and shutdown curves given in the safety report, during an operation at fresh steam pressure, an operation at lower fresh steam temperature is provided for about half an hour. In this operational condition, the total outflow rate can increase up to 15 % according to our estimates. This increase is covered by the conservative estimate of 600 Kg of H_2O of the total mass penetrating into the primary circuit.

By using the detection time specified by the applicant for a break in the heating pipe and the positioning time of the valves, we can confirm that, via the leak of the heating pipe up to the pressure compensation, approx. 170 Kg of water and stream pass into the primary circuit.

The assumption that, after the pressure compensation on the primary side, the entire water inventory passes to the secondary side is conservative. Of decisive importance for the size of the secondary side mass of water at pressure compensation, is the time at which the relief valves were closed. Our estimates confirm that, with the anticipated pipe and valve profiles, a steam generator relief can occur within approx. 40 sec. Herein, only a relief strand is considered, which is in agreement with the assumption of the applicant.

In order to limit the water mass, which can pass into the primary circuit, it is necessary to extract all the water present in liquid form via the discharge lines. The applicant's calculation for the full-load case shows that, at the time of the pressure compensation, only about 300 Kg of steam are present in the secondary volumes restricted by the check valves of the steam generator. We confirm this calculation. Thus, when there is a break in the heating pipe at full-load, a total of 470 Kg of water and steam pass into the primary circuit

Parameter variations show, however, that as a result of the changed starting conditions (for example, the initial pressure), relatively minor amounts of water in liquid state can remain in the feed water collector after the closing of the relief valve at pressure compensation. These amounts are low with respect to the steam generator volume, but, under the assumption that the entire water inventory of the secondary side passes into the primary circuit, they are of influence on the total penetrating mass.

An additional influence variable is the delay time for the closing of the check valves on the side of the fresh steam. That is why, after establishing the construction design of discharge lines and valves, the pressure safeguard of the tension release mechanism and of the relief tanks requires investigations regarding the procedures concerning the relief process. The mentioned parameter variations show, however, that with a delay-free closing of the relief valves, less than 600 Kg of water escape from the heating pipe leak.

It can be verified that, in order to limit the mass that passes into the primary circuit to 600 Kg at full-load, the anticipated concept is suitable for controlling a break in the heating pipe.

Summarizing Evaluation

In the approval procedure for pressurized water reactors, a double-ended break of a heating pipe is examined for possible damage to the steam generator heating pipes. In this process, the applicant follows the assumption of a double-ended heating pipe break. The ramifications of the leaks in the operational steam generator are discussed with respect to the effects of a break in a 2F heating pipe.

When there is a break in a heating pipe, the circuit temperature rises slowly due to the absence of the secondary side heat exhaust due to DE blockage and relief, so that enough time remains to remove the penetrating water from the primary circuit. The maximum temperature of the fuel element remains below its value of approx. 1,300 °C. It remains therefore well below 1,620 °C. The design temperatures of the important technical safety components are not exceeded.

In the analyses for the determination of the maximum core temperatures, the most unfavorable starting conditions as well as erroneous bandwidths of the incoming data were taken into consideration.

For the breakdowns discussed in this section, the applicant assumes the non-availability of all breakdown-reducing operational systems (the blower valve, the breakdown water separator, the pressure regulation, and the helium purification plant).

Only the stimulus criterion is available when there is a break in the heating pipe for registering moisture in the primary circuit and therefore for initiating safety actions (moisture in the primary circuit = 800 vpm). Because of the higher ranked execution of the leading technical devices for the determination of this criterion in the sense of the KTA Rule 3501, we accept the introduction of safety measures in the case of a heating pipe leak via a stimulus criterion.

The moisture level in the primary circuit caused by a heating pipe leak depends on the size of the leak and the resulting amount of water that passes into the primary circuit. The following leakage sizes are examined:

- leaks, which do not address the moisture limit values;
- leaks, which cause the moisture limit values to trigger the reactor safety measures.

Small leaks of the secondary circuit (1 g/s), which do not address the stimulus criterion of the moisture measurement (= 800 vpm), are regulated via the pressure regulator. They are recognized via the expected daily sample taking. In our opinion, these leaks do not represent a danger to the safety of the plant.

With larger leaks, which cause the moisture threshold limits to trigger the reactor safety measures, the amount of water, which penetrates the steam generator and thus passes through the leak into the primary circuit is limited.

These features of the HTR module lead to the fact that, with the design disruption "2F break of a steam generator heating pipe," and with the additional failure of the operational systems such as the breakdown water separator, the helium purification plant, the pressure regulation, and the blower valve, the pressure relief system of the primary circuit is only addressed after 3.5 hours. This is equal to an average speed of pressure increase of approx. 50 mbar/min. The stimulus criterion for the gradient of the primary circuit pressure was selected at 180 mbar/min so that the pressure oscillations due to the operational temperature changes do not lead to an initiation of safety measures.

The water gas production due to corrosion of the fuel elements and the graphite built-in components depends on the amount of water mass penetrating into the primary circuit due to a 2F heating pipe break. We confirm that, in the least favorable case, a total of 600 Kg of water passes into the primary circuit.

The corrosion rates were determined under the assumption that no operational measures for the reduction of the graphite corrosion are in place. Of the calculated maximum corrosion rate of approx. 3.5 %, only a few percentages of the fuel elements, whose fuel-free shell is minimally reduced, are affected.

The determined maximum portion of water gas of 9 % lies well below the permissible values and below the permitted values known to us from approval processes.

The emission of radioactive materials in this disruption is discussed in Chap. 5.8.1.

5.4.4 Load Effects of Cooling Agent Loss Disruptions

According to the rules and regulations listed in Chapter 5.1, the parts provided for the effects of a postulated failure of pressure-sensitive components should be installed as measures for controlling the cooling agent loss disruptions in as much as this is necessary for the warranty of the technical safety security objectives.

The cooling agent, under high pressure and high temperature, causes various load types at the moment of exit from the break opening and affects the surrounding building structures and components.

The leaking cooling agent causes a fast increase of the pressure and the temperature in the ambient atmosphere at the breaking point. The pressure and temperature expand over the existing overflow openings to the neighboring rooms until finally to all building areas, which are connected to one another by flow paths, are included in the pressure compensation measures. As a result of the flow conditions, pressure differences are formed between the individual rooms, which stress the building structures. The atmospheric temperatures adjust relative to the mixing of the cooling agent with the existing colder ambient air. This leads to thermal loads on the building structures and components. When there are breaks in the water-conducting systems result effects due to overflow and moisture penetration.

The cooling agent exiting the break opening as a free stream generates jet forces in addition to temperature influences which impinge on neighboring building sections, which acts in the form of a pressure profile over the impinged surface.

The jet of the cooling agent produces reaction forces at the same time, which act as pushing forces on the broken pipeline or components. At the beginning of the outflow, negative pressure waves run into the affected pipe line system. They lead to transient pressure wave forces at the individual pipeline sections and components as well as their built-in components, which are limited by elbows. These mechanical loads have to be securely discharged into the building via the support structures and equipment safeguards of the systems and their anchors.

In addition, loads also occur in failing containers with large energy content in the form of bursting pressure waves as well as by flying container parts and broken pieces as debris loads, which are to be taken into consideration in the design of the important technically safe buildings.

According to the safety report, the applicant intends to undertake the design of the building structures and components to counteract the effects of the disruptions mentioned above by means of pressure containing components as far as this is relevant to ensure the safety objectives. At the same time, the assumed break postulate and its effects on the quality and energy content of the respective components will be examined relative to:

- In high energy pipelines, that is, with an operational pressure greater than/equal to 20 bar and/or an operational temperature greater than/equal to 100 °C, arbitrary positional circle sketches and below-critical cracks are postulated and the local effects
  - jet forces,
  - reactions forces,
  - pressure wave forces,

- Forces from knocking lines (as a result of reaction forces),

as well as the global effects of

- pressure buildup, pressure differences (for water/steam conducting pipelines with an average temperature greater than/equal to 100 °C),
- temperature (at average temperatures greater than 100°C),
- moisture (in water/steam conducting pipelines with an average temperature greater than 100 °C),
- overflows (in water conducting pipelines).

In not so high-energy pipelines are merely postulated below-critical cracks and, if needed, the effects of the resulting overflows are considered.

As safety measure against the effects of a postulated failure of high-energy pressure conducting components is provided the principle of spatial separation of the relevant technical safety systems, which should be protected from one another, the reactor building, and the reactor building additions, whereby the security function is ensured via the design of the building structures to prevent the local and global load effects.

In addition, there follows a design of the systems and components including the built-in components and fasteners directly against these loads for the prevention of unauthorized expansion of the damage.

For the reactor ancillary buildings exist technical safety protection objectives only with respect to the stability and the tightness against the penetration of radioactive materials into the soil. The design of the buildings considers the globally acting loads due to a pressure build-up, that is, due to pressure differences as a result of the postulated failure of the high-energy pipelines for the warranty of these safety goals.

We agree with the applicant's assumptions with respect to the postulated failure of the pressure conducting components and the dynamic thermal fluid load effects. The anticipated design concept for these loads is compared basically to the already approved and built nuclear energy plants. For the required load evidence calculations are available suitable processes and calculation programs. According to experience, the extent of such loads is conceptually not relevant since a required design of the building structures and components is possible with the corresponding expense or in an individual case, load reduction measures could be considered.

The loads from the effects of the failure of the pressure conducting components form the basis for the so-called special loads due to the effects from within (EVI), and from outside (EVA), and rank in part higher than the operational and regulation loads, and form the basis for the design of the building structures and components. We consider it to be reasonable, in view of the detailed construction planning, to collect the individual EVI-load cases by systems, that is, to catalog the components and load types by category in a special load catalog, as is customary at other nuclear energy plants. Therein are disclosed the basic peripheral conditions, for example, the thermodynamic initial conditions and the safety increases, as well as the calculation processes, which are used.

We do not have any objections with respect to the concept of the control of the load effects due to the failure of the pressure conducting components as presented by the applicant.

Chapter 5.5.6 addresses the effects of a postulated failure of high-energy containers and other components.

Pressure Surge Loads

Flow processes in the pipelines, which drastically change over time, as they occur with the acceleration of the cooling agent due to a break in the pipeline, or via their delay due to a fast closure of a check valve, lead to steep pressure transients, which expand with the speed of sound of the medium in the system, and exert thereby pressure wave forces on the affected components. These forces are also produced, for example, with the opening and closing of safety valves, the closing of the vacuum breakers, and other blockage elements, in pressure relief processes, refill processes, and quick transitions between gaseous and liquid media, in addition to the already considered breaks in pressure conducting components. These operational and disruption processes events occurring at the plants must be investigated and the resulting loads considered in the design of the components and, if necessary, the building structures. These load incidences are normally contained in the special load catalog mentioned above and are analyzed in detail in the context of the construction planning.

Loads on the Reactor Building Caused by Pressure Differences

The concept of the applicant anticipates the implementation of the pressure relief of the reactor building via a direct exhaust of the cooling agent through the chimney into the outdoors, in the case of a cooling agent leakage with greater outflow rates than can be controlled by the capacity of the secured pressurization.

The considered flow paths of the cooling agent through the building lead from the potential room leaks, which contain highly pressurized pipelines, over the existing overflow cross sections, through the rooms of the module plate, upwards up to a height of 12.7 m. At a pressure of 1.05 bar, valves open to a relief shaft, which ends in the reactor hall. When the pressure is raised above 1.1 bar in the reactor hall, valves open to the chimney and open the pressure relief path to the outer atmosphere. The reactor hall with its large partial volumes, consisting of the reactor floor, the steam generator floor, the maintenance shaft, truck entrance, and the connecting paths, thus has the function of a temporary buffer volume in which a limited global building pressure increase takes place. This also includes the two stairways and the second module, which is not affected by the break, at which [air] is directed via valves in the reactor hall.

The applicant presented extensive calculations for the loads of the reactor building after pipeline breaks /U 5.4-14/. In them, the largest anticipated primary and secondary side breaks for the energy relief in the building and the resulting pressure and temperature processes in the individual rooms are addressed.

The manufacturer of the mathematical program PSS 3 calculated the flow rates as well as the building loads. The program calculated the thermodynamic fluid processes in a network of control volumes and overflow connections with the aid of a finite-differential process /U5.4-14/. The program enables the description of the condition of gas, steam and water as well as mixtures thereof under the assumption of a thermodynamic equilibrium. The calculation models for the flow and the heat transfer correspond to the known references in the technical literature and are taken in large part from the VDI Heat Atlas /L 143/. For verification, the applicant provides a number of his own application examples as supplementary information /L 155/ for which he has signed a publication release.

To determine the energy emission rates from the breaks of the high pressure systems in the buildings, the applicant has examined the primary side break of

- a 2F break in the external pressure compensation line at the steam generator pressure vessel,
- a 2F break in the fuel element break line at the break separator,

as well as the secondary side break locations

- a 2F break in the fresh steam line at the fresh steam nozzle,
- a 2F break in the feed water line at the feed water nozzle.

A zone model of the entire primary circuit, which considers the basic geometric data (volumes, flow cross section) was created for the primary side breaks. The data for the applied flow resistors and the heat transfer boundary conditions are documented in the technical supplement /U 5.4-14/. The secondary side of the steam generator was taken as the heat source drain and the nominal operating data were presented as starting conditions. A spontaneous 2F break was assumed as a triggering incident and the reactor safety actions, which occur after a primary side break, were considered for the transients. The two completed break analyses cover the largest primary side connecting rated width (DN 65). In the break of the fuel element break line (DN 125) results an effective outflow cross section, just as with a break of a DN 65 line because of the cross section reduction in the isolation panel.

For the analysis of the secondary side breaks was created a zone model of the steam generator, which illustrates the primary and secondary side volumes and cross sections of the flows, including the flow resistors and heat transfer conditions. The entering and exiting primary side gas mass flows and the secondary side water, that is, the steam mass flows were considered as boundary conditions related to the stationary operation. A spontaneous 2F break was assumed for the fresh steam line, which however leads only to a one-sided, non-blockable outflow from the steam generator, since it was assumed that the break end of the second module was blocked by an overflow valve (see also Chap. 5.4.2). For the 2F break of the feed water line the discharge from the break end on the pump side was neglected due to the relatively minor energy input from this side.

The break mass flows and specific enthalpies of the cooling agent were given as results of the undertaken analyses, and are also used for the calculation of the building loads. The comparison shows, among other things, that the energy emission from the break of the pressure compensation line is covered by the additionally examined break location at the RDB because of the break location at the steam generator. Furthermore, the flow rates at the fresh steam line break clearly cover those at the feed water line break.

The building loads resulting from the individual break incidences were determined in separate calculations. For this purpose, the reactor building was divided into 260 separate volume elements and the energy flow of the break analyses for each representative zone of the respective room break in the building entered into the calculation. The openings existing between the rooms in the building were represented as overflow cross sections with their corresponding pressure loss coefficients and, as far as available, the opening pressures for the valves and rupture film. The heat transfer between the fluid and the building walls was neglected. The load analysis for the feed water line break was waived since the energy input was clearly covered by the fresh steam line break. In each case, the results of the building calculations show the steep increase to be expected in pressure and temperature in the break rooms, which rapidly declines because of the occurring relief due to the exhaust into the neighboring rooms.

For the remaining rooms along the pressure relief path, these processes are more or less time-dependent and dampened in dependence upon the inflow and outflow conditions, the volumes, and possibly the valve opening conditions. With the temporary subsiding of the exhaust rate and after the opening of the pressure relief valve to the chimney, there is a decline in the overflow pressures, and thus also the pressure differences in the building. The temperatures follow this trend qualitatively, but at times heavily time-delayed, since they are dependent upon the structures in the slow heat transfer processes.

The maximum pressure differences during the course of the flow processes as well as between the rooms of the building and the external atmosphere are established as relevant loads for ceilings and walls.

They are by nature locally different in size and depend, among other things, on the break location and the energy release. The maximum pressures differences for the reactor building result in large part from the break in the fresh steam line. The highest value of 1.25 bar of the pressure difference occurs between the break room and its neighboring rooms. The exterior wall of the reactor building and the wall of the primary cell are locally affected by this load. A maximum pressure difference to the outer atmosphere of 0.12 bar occurs for the largest surface area of the exterior walls of the reactor building. For this purpose, the applicant established a design value of 0.3 bar.

The applicant consolidated the totality of the pressure differences resulting from the examined cooling agent loss incidences for the individual room areas in a "matrix for room pressures" /U 5.4-15/. Aside from the mentioned value, no design values for the remaining rooms of the reactor building are given. This should take place within the framework of the construction planning.

We tested the investigations presented by the applicant with respect to the pipeline breaks /U5.4-14/. We agree with the representative model of the design, the design data, and the applied calculation parameters used in the process of analysis. We consider the applied methods of calculation as appropriate for the investigation of the physical processes examined here.

To check the calculation results of the applicant, we carried out comparative calculations with our own thermo-hydraulic computation programs/U 5.4-14/. The model design, as well as the relevant physical computation parameters, were modified, and their influence was evaluated with respect to the result.

The results of our calculations show a satisfactory agreement with those of the applicant. With respect to the 0.3 bar design values for the reactor building, there are a few reserves of an additional 0.1 bar as safety increase which is to be used according to the RSK-guidelines /L 10/.

Based on our tests, we agree with the values of 0.3 bar design pressures for the reactor buildings determined by the applicant, with the restriction that this value refers to the pressure in the reactor hall and applies to the largest surface area of the exterior walls of the reactor building. Other building sections located close to the break experience in part considerably higher break differences and have to be considered individually in the statistical design of the building, and must be taken into consideration as special loads, together with other load types of the loads collective for the design of the building structures.

We are in agreement with the load of the reactor building through pressure differences with respect to the anticipated concept of pressure relief of the reactor building for the containment of the cooling agent loss disruptions.

The applicant retroactively revised his concept of the fresh steam termination for the containment of the building load in the case of a fresh steam pipeline break in the newly presented supplements /U 2.9-1/ and /U 2.9-5/ (also see Chap. 5.4.2). Instead of the valve solution with a combined check/double valve solution on which the above analyses are based, two own medium-driven rapid closure valves are provided, which will be controlled directly by the signal "negative sliding threshold value of the FD pressure."

In this way, a break in the flow from both break ends is to be considered in the case of an FDL break in front of the FD check valves until the valves are closed.

The applicant has compared the pressure load of the reactor building /U 2.9-6/ with the earlier analysis up to an assumed time of closure of the valves with a 1.4 sec. increase of the steam release. Within a short term, this results in higher maximum pressure values in the sheer panel module, while the pressure remains limited to approx. 1.2 bar in the reactor hall and with it continues to lie below the given design pressures of 1.3 bar. The applicant further anticipates reducing the building pressure load via refined calculated methods of proof or via construction measures applied to the overflow cross sections.

We do not object to the changed fresh steam check valves and the consequently resulting higher building loads due to an FDL break in the context of the concept evaluation. Also in accordance with the building pressure calculations tested by us resulted already local high room pressures as 1.3 bar in the module plate, which are to be considered in the building design during the course of further construction planning.

5.5 Additional Internal Design Disruptions

5.5.1 Operational Transients with Assumed Failure of the Rapid Shutdown System (ATWS)

Within the framework of a risk minimization, the RSK guidelines require the examination of certain operational transients with assumed failure of the rapid shutdown system for pressurized water reactors. According to these, the design limits of the pressure conducting enclosure should not be exceeded. Beyond that, the after-heat has to be sufficiently exhausted and the reactor must be able to be turned off. In the analysis of these incidences, all the other systems are assumed to be functioning, with the exception of the presumed failure of the rapid shutdown system, as long as their functionality is not impaired by aftereffects.

With the HTR module reactor, the core is constructed in such a way that the reactor turns itself off because of the temperature increase after the shutdown of the blower due to the negative temperature coefficient.

The applicant performed an examination in particular with reference to the HTR module reactor based on the operational transients mentioned in the RSK guidelines /U 5.5-1/. According to these, the three operational transients to be examined can be divided into three categories. The first category includes all operational transients, which lead to a failure of the main heat drain. The second category includes the transients with the maximum reactivity supply, and the third category counts the pressure relief.

In the absence of main heat drain with a simultaneously assumed failure of the rapid shutdown system, the power reduction is delayed in time because of the negative temperature coefficient. Because of this, the temperature of the central fuel element increases at the beginning by approx. 150 K above the temperature reached with a functioning rapid shutdown system. This temperature increase reduces itself substantially by temperature compensation within a few hours. The temperature loads of the core fixtures are unimportant for the technical safety. Significant consequences on the metallic structures do not occur /U5.5-1/.

The maximum reactivity supply is reached via a removal of all the reflector rods at maximum speed. The applicant examined this case under the assumption that, after a long-term partial load operation, the installation is again operated at full load shortly before the appearance of the transients. The maximum fuel element temperatures increase to approx. 1300 °C. The reactor is shut down via the tripping of the KLAK shutdown devices and the plant is operationally shutdown, so that the core temperature does not rise substantially and clearly remains under 1,620 °C.

The pressure relief due to the unintended opening of the primary safety valve leads, in supposed failure of the closing function, to a total pressure loss in the primary system. The consequences of these transients are comparable to the design interruptions discussed in Chapter 5.4.1.3.

In summary, we note that the risk reduction with a failure of the rapid shutdown system in operational transients required by the RSK guidelines was absolutely provided.

5.5.2 Disruptions in the Handling and Storage of the Fuel Elements

The following events are included among the disruptions in the handling of fuel elements:

- a break of the pellet feed pipe within the RDB,
- disruptions in the fuel element supply and exhaust installation,
- a fall of a fuel element transport container,
- an erroneous sorting of the fuel element transport containers in the warehouse,
- a flooding of a fuel element warehouse,
- influences from outside on the components of the fuel element handling and storage,
- a mix-up of the fuel element transport containers at the time of reloading.

In the following are listed disruption-specific evaluation criteria derived from the rules and regulations listed in Chapter 5.1:

- An improper release of radioactive materials from irradiated fuel elements during a disruption of the handling is to be avoided.
- A reliable after-heat exhaust in the case of a disruption should be ensured
- The installations for the handling and storage have to be such that a critical disruption and disallowed radiation exposure can be ruled out.

Break of a Pellet Supply Pipe within the RDB

With an undetected break of the central intake pipe within the RDB, the fuel elements can reach into the annular area between the core container and the core built-in components and into the floor calotte of the reactor pressure vessel. The fuel element supply pipe is located in the cold gas chamber and is loaded with a negligible excess pressure. The detection of the affected fuel elements occurs via a counter and via the monitoring of the pressure and the flow of the carrier gas. When there is damage to the fuel element supply pipe and at the conveying controls, a manual shutdown of the plant takes place /U 2.5.5-9).

We consider an accumulation of a larger amount of fuel elements in the reactor pressure chamber outside of the core to be impossible. Beyond this, the buildup of a critical accumulation outside of the core in the RDB based on spatial conditions is not possible.

Disruptions of the Fuel Element Feed and Exhaust Plant

Disruptions in these installations are possible via postulated breaks in the pipelines and operational mistakes during the infiltration and exfiltration of the fuel elements.

In as much as they cannot be stopped, these breaks lead to a pressure reduction in the primary circuit in the least favorable case. Chapter 5.4.1.1 discusses this type of disruption.

We consider it a requirement to develop a concept for supervision of the fuel element feed and exhaust plant during the construction planning of this plant, which will reliably prevent the improper handling through the installation of a locking system. In this case, the experiences with the THTR and AVR are to be taken into account.

Fall of a Fuel Element Transport Container

Fresh fuel elements are transported and stored in containers on wheels. The same containers are used in the THTR. They have two-layered walls and can hold up to 1,000 fuel elements. The layer between the inner and outer wall is filled with an absorbent. According to our opinion, the low criticality of the integrity loss due to a fall, when there is a loss of absorbent, is ensured by the limited number of fuel elements.

The combusted fuel elements are temporarily stored and transported in transport and storage containers constructed specifically for this purpose, whose approval is as yet pending. Similar containers were developed for the DWR fuel elements. The expected size of the container is for 30,000 to 45,000 fuel elements. The transport and storage containers should be designed in such a way that the integrity and thus the residual radioactive materials ensure the thermal exhaust and low criticality, even in the case of a fall of the container. This is also applies for the containers for holding pieces of broken fuel element.

Incorrect Sorting of Fuel Element Transport and Storage Containers in the Warehouse

The containers with fresh fuel elements are stored in the reactor ancillary building or the reactor building. Up to 4 containers can be stacked in the reactor ancillary building. In view of the assurance of low criticality, an arbitrary arrangement and storage is possible due to the existing absorbents lying between the inner and outer walls of the containers.

The containers for the combusted fuel elements and the containers for the pieces of broken fuel element are stored in so-called intermediate storage. These containers are positioned vertically standing one next to the other. From the construction standpoint, a stacking is precluded because of the limited height of the building. Because of the design and regulations for these containers, a low criticality and after-heat exhaust in the temporary storage are ensured.

Flooding of the Fuel Element Storage

A flooding of the storage areas for the fresh or combusted fuel elements could lead to a change of the neutron moderation and reflection. This has to be considered in the design of the fuel element storage areas for the temporary storage in the reactor ancillary building or the reactor building.

External Influences on the Components of the Fuel Element Handling and Storage

The fuel element transport containers and the storage areas for the fresh fuel elements are not designed to withstand external influences such as earthquakes, pressure waves, or an airplane crash. In our opinion, we do not consider such a design a requirement, since at such occurrences it can be ruled out that enough containers would be arranged in a favorable manner for the criticality, would open, fill with water, and lose the absorbent.

The fuel element transport containers for combusted fuel elements are designed for external influences such as earthquakes, pressure waves, and an airplane crash. A developing critical formation or emission of radioactive materials as a result of such events can thus be ruled out.

Mix-up of the Transport Containers during a Reloading

In a possible core unloading, the necessary transport containers are marked for identification. Hereby, a mix-up during the reloading can be administratively ruled out. We agree with this concept. This identification system has to be described in detail in the operations manual.

Summarizing Evaluation

In summary, we have noted that the consequences of a postulated disruption during the fuel element handling and storage are controlled. The anticipated measures and regulations are suitable to prevent during a disruption of the fuel element handling an unauthorized emission of radioactive materials to ensure a low criticality. The evidence of safeguarding of the low criticality and after-heat exhaust in the fuel element storage must take place during the construction planning of the plant.

5.5.3 Fall of Heavy Loads

In addition to the handling of the fuel element transport containers, concrete locks on the primary cells, RDB-covers, and the primary circuit blowers are undertaken within the reactor building in accordance with the increased requirements of the design of the reactor building crane following KTA Regulation 3902 / 67/. The weights of these components are similar to the weight of the fuel element transport containers.

The opening of the RDB, and thus a transport of the RDB cover at the time of WKP work on the drives of the shutdown devices, as well as when there is damage or repairs are carried out on the RDB interior built-in components is necessary. The expansion and transport of the primary circuit blower is only required when there is substantial blower damage. In both cases, the cement locks of the primary cells also have to be removed.

Transports of heavy loads only take place during a switched-off cold low-critical condition of the reactor. We are of the opinion, taking into consideration the design of the reactor building crane according to KTA Rule 3902 as well as administrative measures, careful inspections, and functionality tests, that a crash of these loads can be precluded.

5.5.4 Disruptions of the Ventilation System

We consider disruptions and failures of the individual plant parts of the safeguarded underpressure containment as substantial disruptions in the ventilation system. The safeguarded pressure containment has the function, in addition to the pressure relief of the primary circuit with following core heating, of releasing the radioactive materials emitted by the primary circuit in the reactor building filtered via the chimney into the environment.

With disruptions to the exhaust system, which lead to the failure of the secured pressure containment, the filter for suspended particles of the continuous air filtering system is available for filtering. Because of the increased radiation exposure in the environment, it remains even then below the threshold limit according to § 28 (3) of the Radiation Protection Regulation.

5.5.5 Disruptions in the Helium Purification Plant

The helium purification plant has the operational function of removing dust and gaseous contaminants from the primary cooling agent of both primary circuits. In the case of steam generator leaks, the helium purification plant removes the invading water and possible corrosion products from the primary circuit.

During the investigation of steam generator leaks, also the failure of the helium purification plant was taken into consideration, including the pressure regulation and the water separator. In this case also, the steam generator leaks are manageable.

We thus conclude that it is technically safe to accept a short-term failure or an unavailability of the helium purification plant. Over a long term, the operation of the helium purification plant is required for the separation of contaminants or the moisture that may have possibly penetrated.

Breaks or leaks of the primary cooling agent carrying lines of the helium purification plant are discussed in Chapter 5.4.1.3.

5.5.6 Generation of Projectiles

Projectiles are parts resulting from breaks, which can be generated by the failure of the pressure carrying components or the failure of components with rotating parts.

For protection against the failure of the components mentioned above, the applicant anticipates the following measures:

- a spatial separation or suitable spatial arrangement
- a design against the resulting loads
- protective or containing constructions

The safety report discloses that the pressure carrying components and components with rotating parts are designed to withstand all of the expected requirements. These components are to be made of a material, which is suitable for the purpose, and are manufactured, mounted, assembled, tested, and operated under a comprehensive quality assurance.

Beyond this, the failure of the pressure carrying components and components with rotating parts are the basis for the design of the structure and its components, when the consequences of the failure are relevant for the compliance of the protection goals.

In this paragraph, the pressure carrying components are understood as containers with great energy content and cable entries. According to the safety report, the bursting pressure wave and broken pieces as a consequence of the container failure are to be considered in the design of the technically relevant safety structures and in their design parts. The broken fuel element tankard is also considered among the containers with great energy content. The tankard is located in a transport container, which is installed in the case of a bursting pressure wave due to a failure of the break tankard. The pressure vessel unit (DBE) is the only exception where a failure is not presumed. Because of restraint construction, possible consequences to the cooling area because of the failure of the cable entries at the DBE are ruled out.

Water supply containers and clean gas storage containers are located in the machine housing or in the gas supply plant. They should be regulated in such a way that, when they fail, the important technical safety-relevant buildings for the containers for the broken pieces cannot be affected.

Broken pieces of components with rotating parts can occur due to a failure of the turbines, pump lines, compressors, blowers, ventilation, E-motors, and generators. If a failure of these components cannot be ruled out, and important technical safety-relevant devices can be provided at the same time, a protection occurs in such a way, that possible broken pieces cannot penetrate into the structure of the affected components. In turbines, the alignment of keel line to the reactor building serves as an alternative measure.

We agree with the concept for controlling the effects of the failure of pressure carrying components and those with rotating parts. The load assumptions in the construction planning of the respective buildings and their components have to determined and tested.

5.5.7 Internal Plant Fires and Explosions

The measures for protection against fire and explosions are discussed in the fire protection assessment. For this purpose, we provided supplementary data in Chapter 2.15. According to our opinion, requirements beyond this should not be raised in view of the disruption analysis within the context of the conceptual evaluation.

5.5.8 Disruptions of the Structural Regulations

To the operation of the plant belongs the regulation of the following variables:

- the thermal performance,
- the fresh steam and processing steam temperature,
- the hot gas temperature,
- the fresh stream and processing steam pressure,
- the water supply container pressure,
- the turbine counterpressure,.
- the turbine revolutions.

Transients can be triggered by the malfunction of the control devices and, in the case of a disruption, each one of these variables can take on an unfavorable value.

The ramifications of the disruptions of the control devices were discussed in the previous Chapters 5.2 to 5.4. A failure of operational control is generally presumed in the investigations of disruption incidences. A detailed testimony about the influence of the operational control systems on the disruption processes can only be deduced in the course of the construction planning.

5.5.9 Internal Structural Flooding

An internal plant flooding can occur when the water carrying systems (for example, surface coolers, blower coolers) fail.

Measures against water damage (for example, of components, realignment of cables and pipelines) must be established during the construction planning of the HTR module. Within the frame of the assessment, it is our opinion that no detailed treatment of measures against an internal plant flooding is required.

5.6 External Influences

This chapter considers the natural and civilization-dependent external influences, which should be, due to their potential of endangerment, based on the scientific and technical state of the nuclear power plant design. To these events belong:

- Earthquakes,
- Floods,
- intake of poisonous, explosive and corrosive gases,
- fire in the surrounding area,
- debris.

In addition, protective measures against airplane crashes and pressure waves from chemical explosions as external effects with very low frequency of occurrence are met for risk minimization.

Chapter 2.4, "Building Structures," addresses the influences of storm, hail, and ice formation. Chapter 2.12, "Electrical Plants," addresses lightning protection measures in detail. The possibilities and ramifications of further events, such as, for example, underground cave-ins and explosions, cannot be discussed here; they are to be evaluated in dependence upon the location-specific construction site. These results have no influences on the concept of the HTR module or on the construction execution.

The protection of the plant from third party influences also has to be specifically considered. The evaluation of the measures resulting from these is not part of this assessment.

We have based our evaluation of the protective measures of the plant against external influences on Criterion 2.6 of the safety criteria for nuclear power plants /L 6/. From this can be derived the following higher-ranking safety objectives:

- the safe shutdown of the reactor and the continuation of this condition,
- the long-term thermal discharge from the reactor core,
- the prevention of an unauthorized release of radioactive materials.

The following examines to what extent the objectives for protection mentioned above are fulfilled to according to the safety report and the additionally submitted documents for the individual external influences.

5.6.1 Earthquake

According to earlier licensing practices of the Federal Republic of Germany, the nuclear power plants at all locations are to be designed against earthquakes.

In accordance with the safety report, the HTR module power plant anticipates to design the following structures against earthquakes:

- the reactor building (UJA),
- the reactor building annex (UJH),
- the reactor ancillary buildings (UKA)*,
- the building well of the reactor auxiliary plant buildings,
- the switching station and emergency supply building (UBR),
- the cable channels (from UBR to UJA),
- the safeguarded wet cooling cells (URB)

According to the safety report, besides the design of these structures, it is the intention design those building parts necessary for safety in case of an earthquake in order to achieve the above mentioned safety goals. In the following are listed the basic design parts:

* Designed for structural safety based on DIN 4149 with the location-dependent intensity of earthquake safety

- the pressure containment unit,
- the devices in the reactor pressure vessel for the shutdown and the long-term sustainability of low-criticality,
- the primary gas enclosure in the reactor building up to the check valves,
- the devices for fuel element break elimination,
- the steam generator, including heating pipe,
- the fresh steam and water supply pipelines in the reactor building *,
- the safeguarded temporary cooling system of the required supply, information, and control devices,
- the reactor safety system,
- the control room,
- the emergency power plant, including the auxiliary systems,
- the safeguarded secondary water cooling system
- devices in the primary cell, which could influence the function of Class I components (for example, the surface cooler)
- the transport and storage containers.

The design takes place according to KTA-Regulation 2201 /L 41/ for two measurable earthquakes, the safety earthquake (SEB), and the design earthquake (AEB). In reference with the load assumptions, appropriate current scientific references of location-specific surveys of geology, seismography, and building ground are specified. With the consideration of newer scientific results, this will proceed according to the following principles:

- the determination of the site intensities based on earthquake frequency,

* The integrity and function of the check valves is ensured

- the determination of the free-field response spectra depending on the intensity and the ground for the two horizontal ground accelerations as average value spectra (50 %-fractals),
- the vertical dimensioning spectra is defined via 2/3 of the horizontal component spectra and is superimposed on these,
- the strong vibration durations should be chosen as intensity independent but soil dependent

Chapter 1 of the safety report lists concrete resulting load specifications relating to this process.

The choice of the structures against earthquakes and the important technical safety building parts to be designed is basically suitable for the achievement of the safety goals mentioned in the beginning. In relation to the relevancy of the concept evaluation, we pointed out additional areas of design in the chapters of the affected building sections. At the time of the detailed construction planning, the necessity of the design of further structural sections could occur, however in limited scope. In Chapter 5.2.6 we discussed the effects of the compression of the pebble bed by earthquake vibrations. Chapter 5.8 deals with the radiological affects of an earthquake on the environment.

The load assumptions required as evidence of the design against earthquakes are described via the seismic engineering characteristics and are basically:

- the maximum ground acceleration,
- the free-field response spectrum,
- the strong vibration duration,
  or alternatively via
- the elapsed time,

The response spectrum and the time lapse are generally used today in calculation processes of loads due to an earthquake.

Together with the dynamic characteristics of the site ground, the seismic engineering characteristics determine the demands on a structure and its components by an earthquake. The dynamic soil parameters as well as the seismic engineering are determined via the location-specific scientific surveys of seismology or the site ground on the basis KTA-Regulation 2201 for the purpose of obtaining a license.

In the following comparison we consider the orientating approach of the still valid KTA-Regulation 2201 and the newer methods of a realistic description of earthquake loads forming the basis of the earthquake design for the HTR module power plant design. From the comparison of these two methods we derive our evaluation of the intended process according to the safety reports for the determination of seismic load assumptions. The concrete seismic engineering characteristics are evaluated in Chapter 1 "Location."

The determination of the seismic load assumptions is currently based on a pure deterministic approach. Starting with the location intensity, the American free-field standard response spectrum (USAEC, Regulatory Guide 1.60) forms the basis for the dynamic calculation of the modified lower frequency area and the scaled maximum location-specific ground acceleration of the load function.

The basic critical points of this approach are:

- The standard response spectrum is atypical for the German circumstances since it is based on earthquake registrations in seismic active zones, which differ in their macro-seismic vibration parameter markedly from those in areas of lesser seismic activity. Thus, the earthquake magnitudes for German locations are less and the associated range distances are shorter.

- The scaling of the standard spectrum with the maximum ground acceleration, the so-called engagement value, is physically doubtful. Besides, the determination of the maximum acceleration from the location intensity is uncertain, since there are a number of correlations between both parameters, which disperse within wide limits.

- For the dynamic calculation, after the lapsed time process, artificial earthquake curves with forced, unnaturally wide frequency content are generated from the standard response spectrum. In addition, the time processes show a conservative estimated strong vibration duration.

Newer methods for a more realistic description of earthquake loads start from an analysis of the probability of earthquake endangerment at the site and employ statistical empirical methods for the direct determination of seismic engineering characteristics from representative free-field registrations for the site.

With an assumed excess rate of the earthquake intensity, the appropriate site intensity can be taken from a FDR matrix of probable earthquake endangerment. There is an empirical connection between the intensity and the remaining macro-seismic parameters, the magnitude and the distance range. The representative earthquake registrations in reference to these parameters and the underground conditions at a site are statistically evaluated with respect to the seismic engineering parameters. In this way, are obtained the intensity and the subsoil-dependent free-field response spectra and strong vibration duration or natural free-field lapsed time.

Critical comments need to be made on this process:

- With a probability earthquake prediction, the question continually arises concerning a generally accepted measurement for the underlying possibility of exceeding [the prediction]. This applies particularly to the coupling of a lower measure of excess rate (10^{-5}/a) with the average value spectra.

- The observed number of earthquakes is relatively low for areas of lower seismic activity, so that the earthquake prediction will not be very exact on the basis of these data.

- The earthquake statistic assumes that the individual results are independent of one another. This does not apply to seismo-tectonic processes.

- Earthquakes occur more often in spatial and timely concentrations. This limits the results of earthquake predictions using probability, which start with an equal distribution of earthquake activity over a delimited region and with a statistical occurrence frequency or period of recurrence.

In the nearby area of an earthquake, which is relevant to the demands on the structure, should be expected deviations of the expected intensity with respect to the data of the earthquake endangerment matrix because of local geological and seismic-tectonic influences, which have not yet been clarified in all respects.

We derive the following evaluation of the anticipated process listed in the safety report as a factor of the seismic load assumption from the foregoing comparative presentation.

The design of two earthquake measurements – safety earthquake and design earthquake – corresponds to the currently valid regulations; however, newer scientific insights have been found in the meantime for a draft of changes in KTA-Regulation 2201.1. According to this, the safety and design earthquakes will be merged into a single encompassing earthquake measurement. In our opinion, these changes in the KTA-Regulation will be approved and will be followed in the future.

The requirements of the site intensity on the basis of the occurring frequency of certain earthquake strengths permit a good estimate of the earthquake endangerment in the region of an arbitrary site. The intensity assumed in the safety report (see Chap.1) corresponds absolutely to possible site conditions within the Federal Republic of Germany. This process, however, does not replace a site-specific seismic intensity analysis under consideration of local geological and seismo-tectonic influences, as they are generally undertaken in a seismological evaluation.

The current practice of defining measurement spectra in the form of standard free-field spectra with maximum accelerations as scaling value leads to unrealistic load numbers, which however lend themselves to a conservative upper estimate of the structure demands and excitation transfer to the components.

Since an extensive database with representative earthquake registrations for the area of the Federal Republic of Germany already exists and an obligation to include the progressive knowledge base exists, the present deterministic process methods should be compared to the newer methods for a realistic description of the earthquake loads in the form of intensity and subsoil-dependent free-field response spectra and strong vibration duration of free-field lapsed time.

The stated overlapping regulation and the size of the vertical measurement spectra compares to the current state of the technology of component-related data.

In summary, we judge that the processes given in the safety report for determining seismic load assumptions is a consistent process for the realistic description of earthquake loads. In an at least affordable site-specific seismological survey, this process should be used taking into consideration the deterministic methods used up to now in order to determine the final seismic engineering parameters without a loss of safety. In Chapter 1, we address the seismic engineering parameters, which are to be expected as a result of the assumption stated in the safety report for site features.

5.6.2 Airplane Crash and Explosion Pressure Waves

According to the presented conceptual planning, the reactor building and the important technical safety design elements within the reactor building are to be designed against loads from an airplane crash and explosion pressure waves. To these design parts belong:

- the pressure vessel unit,
- the devices within the reactor pressure vessel for the shutdown and a long-term preservation of low-criticality,
- the primary gas enclosure in the reactor building up to the check valves,
- the steam generator including the heating pipe,
- the fresh steam and water supply lines in the reactor building *,
- the secured temporary cooling system **,
- the emergency steering system including the required supply, information, and control devices,
- the reactor safety system within the reactor building,
- the plants in the primary cell, which influence the function of components of Class I (for example, the surface coolers) in the event of an airplane crash or explosion pressure waves,
- the transport and storage containers

In the case of an airplane crash or an explosion pressure wave, the partial or complete destruction of the switching unit or emergency supply building must be assumed. In this way, the function of the reactor protection system, which is not designed to withstand these events and the emergency supply can be affected or can fail.

---

* Integrity and function of the check valve safeguarded

** Surface cooler up to the fire hydrant connections

With the design of the reactor safety system, the conceptual plan ensures that the reactor safety actions of

- descent of the reflector rods,
- shutdown of the primary circuit blower,
- closure of the secondary cycle,
- closure of the primary circuit,
- the steam generator discharge,

are triggered from an occurring necessity or as a result of damages to the reactor safety system (see Chap.2.13). Due to the inherent safety features of the HTR module power plant design in connection with the anticipated design scope addressing the external influences considered here, the safety goals mentioned in the beginning are also achieved in this case.

The monitoring of the plant and the long-term safeguarding of the low criticality take place from the emergency control station, which is located in the reactor building designed for these events. It is freely accessible and has an external entrance.

To ensure the energy supply in the case of such events and to supply the peripheral systems (ventilation, illumination, communication systems), the emergency control station owns a battery operated, single-line emergency electric network. The battery is designed for 15 hours of operation. A mobile emergency electric aggregate can secure the energy supply beyond this time frame until the network supply returns via an existing cable connection supplied by the fire department or other organizations. If necessary, two surface coolers can be supplied via fire department connections to the temporary cooling system. At the earliest, this is necessary after 15 hours.

The security report states as the basis the load assumptions for the airplane crash event in the design of the reactor building and of the important technical safety design sections; they correspond to those of the RSK-LL /L 10/ and are applied in dependence upon the location.

The results of the explosion pressure wave on the design of the reactor building and the important technical safety design sections are based on the standard pressure process provided in the guidelines of the Federal Department of the Interior for the protection of nuclear energy plants against pressure waves from chemical reactions /L 17/. If, because of location-specific peculiarities, for example, because of the closeness to industrial installations, explosion pressure waves with higher loads than the standard pressure process provide in the guideline /L 17/ become physically possible, the design will take place on the basis of the location-independent actual pressure process.

The events airplane crash and explosion pressure waves are external influences of utterly low occurrence frequency. Their possible affects are therefore not classified as design disruption in the disruption guidelines for pressurized water reactors /L 11/. Those required design features act as measures for risk minimization.

We agree with the design of the reactor building and the important technical safety design elements listed in the safety report against airplane crash and explosion pressure waves as well as to the process method for the determination of the load assumption. We also consider them basically as appropriate for the achievement of the safety goals to be observed in these external events.

The reactivity influx due to a closing of the pebble filling via the induced agitation due to high frequency and short-term stimulus is less than in the case of an earthquake and therefore covered by the observations in Chapter 5.2.6.

The possible waiver of a latent heat exhaust for the duration of at least 15 hours in connection with the independent emergency electricity supply of the emergency control station warrant a sufficient self-reliance of the installation, so that measures for the resumption of the operation of the surface coolers is accomplished and the latent heat exhaust is carried out.

5.6.3 Other External Influences

Flooding

The occurrences and effects due to flooding are location-dependent. The flooding of important technical safety structures and a possible resulting function impairment of the important technical safety installations can be excluded because of the intended design of the nuclear energy power plant according to KTS 2201 /L 42/. This also prevents an unauthorized release of radioactive materials due to a flooding of the structures with active inventory.

The intended design is suitable for meeting the earlier mentioned higher-ranking safety goals.

Influence of poisonous, explosive, and corrosive gases

The permeation or aspiration of poisonous gases through the ventilation system can lead to the defection of the operational personnel. According to the safety report statements, the detection of poisonous gases is not anticipated.

As safety measures against poisonous gases and their effects, it is possible to manually trigger a shutdown of the ventilation valves in the reactor, the reactor support building, the switching stations, and the emergency supply building. Beyond that, breathing protection gear with pressurized air supply is available for the control room personnel.

According to the safety report, the permeation or aspiration of explosive gases is detected via a gas warning installation, which triggers a gas alarm. Beyond that, administrative measures can trigger a gas alarm. Independently of this, the entrance openings of the reactor support building can be protected from the permeation of explosive gases via check valves.

Corrosive gases do not cause short-term damages. That is why the possibility of detection is not anticipated. If need be, the ventilation can be manually terminated.

The possibilities of a permeation or aspiration of poisonous, explosive, and corrosive gases depends on the conditions of the location and its environment. We do not assign a decisive importance to these occurrences and the addressed measures. The measures stated in the safety report are basically suitable to achieve the earlier mentioned safety goals.

Fire in the Surrounding Area

Fires, whose origins are external to the plant such as, for example, forest fires, fires due to fuel tankers, or fuel fires after an airplane crash are to be considered as possible external influences.

The protection from the immediate effects of fuel fires on the power plant site belongs to the scope of the design against an airplane crash. The anticipated protection of the plant against explosive gases also covers the possibility of the effect of explosive gases as a result of fires in the surrounding area. Further devices for the protection against fire in the surrounding area are not anticipated.

With fires due to an airplane crash, the following results or side effects can occur:

- the immediate effect of burning fuel on the building with the possibility of permeation through openings,
- heat effects on the exterior walls of the building,
- the aspiration of explosive gases and damaging smoke gases.

In the context of the anticipated design against an airplane crash, the mentioned consequences can be accommodated in the detailed planning of the reactor building.

Fires external to the energy power plant site are viewed according to their type and infrequency as only location-dependent, since they depend on the existing transport paths, forest density, and other environmental conditions. By removing the important technical safety building from the energy power plant fence, one can achieve that only the effect of explosive gases or damaging smoke gases occurs as a relevant load.

The demands resulting from this can be met with the anticipated measures against the aspiration of explosive and poisonous gases.

Debris

To prevent the effects of debris on the important technical safety installations buildings. such as those which themselves do not contain important technical safety installations but contain in their affected areas safety devices, are also to be fixedly installed therein. This principle has been observed in the conceptual planning of the HTR module power plant site. In Chapter 2.4.10, we address the supplementary design of the continuous air chimney, which is necessary in our opinion.

An airplane crash or explosion pressure wave can cause building and wreckage parts und debris pieces to fly around and hit important technical safety structures. The reactor building, which houses the necessary structural elements for the control of an airplane crash or an explosion pressure wave, is protected against these external effects and thus against the lower loads, which can occur because of building or wreckage parts.

5.7 Interaction of the Modules

The HTR 2-module power plant consists of two modules, which can be operated independently of one another /U 5.7-1/. The design ensures that all the reactor-side disruptions (for example, the reactivity disruptions, performance disruptions, leaks of the primary circuit, and the steam generator) of the effected module are shutdown automatically and within seconds, whereas the other one can still remain operational in principle. Each module is basically assigned a nuclear steam generator, a steam consumer, and a reactor safety system. An interaction can be called upon by EVA, which is assigned to both modules; events beyond the structure, for example, the turbine and vessel burst, via a fresh steam pipeline break, and via malfunctions of the systems.

In our opinion, the following requirements are to be particularly noted because of the nature of the 2-module design:

- in the case of a disruption in one module, it must be possible to safely shut down the second module,
- mechanical failures in one module should not cause damage affecting safety in the other module,
- the operational disruptions of one module should not cause disruptions in the second module.

The site plan shows that both turbines are located next to one another. The EVA or structure internal mechanical damages of one of the turbine or vessel failures in the mechanical building can also cause mechanical damage to the secondary circuit of the other module, which can cause the loss of the main heat drain in both modules. These disruptions are controlled in the design (see Chapter 5.3).

According to the currently available experience with power plant failures, the loss of one module does not have such repercussions on the integrated electric network that a continuous operation of the other module is endangered.

The basic and emergency electric supply requirements are based on double cables. The assignment of the operational users was assigned so that, in the case of the loss of one cable of the emergency supply, only one module is affected. This event leads to the shutdown of the affected module.

The intermediate and secondary water-cooling systems are only once present in the plant; they supply the surface coolers of both reactor modules. The double cables of the surface cooler are connected to the secure intermediate water-cooling system and to the secondary water-cooling system; the remaining third cable is connected to the nuclear intermediate water-cooling system. A loss of these systems leads to the inability to cool the surface coolers and the reactor pressure vessel and the steam generator supports. The safeguarded intermediate water-cooling system and the safeguarded secondary water-cooling system are double cabled, supplied by emergency electricity, and designed against earthquakes. In a postulated loss of these systems, the installation can be cold-driven utilizing the main heat transfer system. The reactor pressure vessel and the steam generator supports can by cooled by utilizing the nuclear intermediate and secondary water-cooling systems.

The disruption water separator of the helium purification plant is single cabled. This is needed in order to remove the accumulated water gas and the moisture from the primary circuit in the case of a heating pipe break. Chapter 5.4.3 examines the loss of the disruption water separator in this case.

If a safeguarded underpressure containment at the loss of the single cable is assumed to be caused by a pressure release disruption with attached core heating, the analyses in Chapter 5.8 show that the function of the safeguarded pressure containment is not required for limiting the radiation exposure according to the stated planning threshold values of § 28 (3).

With a fresh steam pipeline break in one module, the effects via the meshing of the secondary side of both modules can be influenced. We address this in Chapter 5.4.2.

5.8 Radiological Effects of the Disruptions

5.8.1 Release of Radioactive Materials

5.8.1.1 Selections of Representative Disruptions

The selection of the disruptions to be analyzed from the radiological viewpoint for nuclear power plants with pressurized water reactors is given in the disruption guidelines /L 14/. These disruption analyses should be based, according to the disruption guidelines, on sufficiently accurate assumptions, calculation models, and parameters for the disruption process, the release and expansion of radioactive materials.

Because of the different design of a pressurized water reactor, other cases of disruption could be design-determining factors for the HTR module (see Chapter 5.1). We examined the selection of the design disruptions, with a reasonable application of the disruption guidelines, as stated by the applicant, to see if the selected cases of disruption are, in this sense, radiological representations and if they cover similarly occurring events in their radiological effects. The applicant selected the following disruption cases:

- A leak in a connection line between the reactor pressure vessel and the primary circuit check valve,
- a leak in a primary cooling agent carrying metering line,
- a failure of a steam generator heating pipe with long-term loss of the water separator and of the primary circuit pressure regulator.

The analyses of these three disruption cases basically serve to prove that, with the release of radioactive materials in the reactor building, the radiation exposure in the environment is sufficiently limited.

Of those disruption cases where the activity is released outside of the reactor building, the applicant investigated the following cases:

- a leak in the largest primary cooling agent carrying pipeline outside of the reactor building and a closing of the primary circuit stop valve,
- a leakage of a container with radioactively contaminated water.

The analysis of these disruption cases serves to prove that the release in the case of disruptions outside of the reactor building is sufficiently limited.

Another analyzed disruption is the earthquake against whose effect the reactor support building site is not designed. Here the applicant assumes

- a leakage of the components of the helium purification plant and at the evaporation concentrate containers.

In the scope of this investigation, we did not consider any additional disruptions since, according to our opinion, the disruptions mentioned above are covered. This is the case, because in all other disruption cases, the released amount of radioactive materials is smaller, the nuclear composition of the released radioactive materials does not differ substantially and the release caused by the disruption into the environment occurs in a similar manner.

5.8.1.2 Leak in a Connecting Pipe Between the Reactor Pressure Vessel and the Primary Circuit Stop Valve

In the safety report, the applicant covered the disruption "break in a large connecting pipe between the pressure vessel unit and the primary circuit closure" and the related radiological effects in a supplementary document /U 3.3-2/. After the break of the pipeline, the primary circuit is emptied within minutes. In this way, the maximum change in the flow rate consists of less than 20% in comparison with the operational requirements. The exhausting primary cooling agent disperses via the overflow opening into the individual sections of the reactor building and reaches the chimney via the pressure reduction valve as a mixture of air and primary cooling agent.

The depressurized primary circuit heats up because of the after-decay heat.

As specifically stated in Chap. 5.4.1.1 for the reactor core, a time and location-dependent temperature load results. The maximum core temperature is reached after approx. 32 hours. The activity release calculations assume a maximum fuel rod temperature of 1620 °C. In addition, supplement /U 3.3-2/ lists the core volume ratio, which at different times exceed specific temperatures. According to this, less than 2% of the fuel rods reach a temperature above 1,500 °C for approximately 40 hours. The location and time dependent temperature distribution is entered into the FRESCO program for the calculation of the release of fission product.

An expansion of the primary gases is associated with the heating of the reactor core. A natural convection to the place of the leak does not occur. The volume difference between the primary gas at operating temperature and maximum fuel rod temperature is released into the reactor building.

The disruption process of a leak in a connection line between the reactor pressure vessel and the primary circuit stop valve is considered in Chapter 5.4.1.1. It also covers other primary circuit leaks regarding the release of radioactive materials.

Pressure Discharge Phase

The applicant considers four activity sources in respect to the activity release /U 3.3-2/:

- the stationary cool gas activity,
- the adsorption of fission products absorptive-dependent on primary circuit surfaces,
- the remobilization of dust-dependent stored activity in dead flow rooms,
- the adsorption of absorptive-dependent activity in the helium purification plant.

The following assumptions were used for the requirement of the activity results:

- a release of the entire stationary cooling agent activity, including the absorptive-dependent inert gas, tritium, and the C14-activity in the helium purification plant.
- the adsorption of the surface activity up to the pressure equalization based on an assumed adsorption-desorption-equilibrium of the partial pressure between the primary cooling agent contamination by radioactive materials and the surface covering with radioactive materials. With this model, the activity released by desorption corresponds to almost 4 times the stationary cooling agent activity,
- the release of 1 Kg of dust with the specific activity for long-lived radio nuclei as in the surface graphite of the fuel rods,
- the neglect of the storage effects on the building and of the remnant of the residual quantities, so that the escape factor of the reactor building results in 100%,

- the chemical formula for radioactive iodine is assumed at 100% as elementary iodine.

/U 3.3-2/ gives nuclei-specific contributions of the individual sources. The summarized activity results from the pressure discharge phase after a break in the connection pipe between the reactor pressure vessel and the primary circuit stop valve, which are provided in Table 5.8.1.1. The releases of the activities given in Table 5.8.1.1 occur unfiltered through the pressure discharge valves of the reactor building and of the reactor support ancillary building in the exhaust air chimney, so that the emission of radioactive materials occurs at a height of 60 m.

The stationary cooling gas activity is appropriately considered in the report /U 3.3-1/ (see Chapter 3.3.1). The releases in the case of a disruption are based on the design value for the stationary cooling agent activity. This is based on the assumed low sediment per primary cooling agent circulation of 10% and thus leads to an order of magnitude higher than one activity concentration in the primary circuit, rather than to the expected value of 90% sediment per primary cooling agent circulation.

For the desorption of the primary circuit surface areas absorptive-dependent fission products, an equilibrium between the activity of the primary gases and the surfaces is assumed, with a distribution factor or ratio factor M, which results in the maximum ratio based on the assumed 32 full-load years of operation.

Since leaks do not lead to erosive surface flows, we consider the model mentioned above to be conservative. According to this model, the partial pressure losses of gas carrying fission products, because of the loss of matter from leak outflows, are spontaneously compensated via a surface non-absorption, so that the partial pressure as well as the relationship of the concentration of radioactive materials in the air and that of surface area contamination remains constant.

The released activity via non-absorption of the surface area can be approximated to a function of the stationary cooling agent activity so that the design values of the cooling agent activity enter into the calculation as a covered case.

The applicant appropriately shows in the report /U3.3-2/ that, in a leak, only a dust release from dead flow rooms near the place of the leak can be expected. No adsorption of this dust by the gas carrying activity is to be expected, so that the beginning of the specific fuel rod surface activity is appropriate with respect to the specific dust activity. The assumption of a dust release of 1 Kg with respect of a dust occurrence of approximately 1 Mg in 32 full-load years and the dust filtering in the helium purification plant are covered.

The maximum activity in the helium purification plant has been appropriately investigated in the report /U 3.3-1/. The design values are conservative in nature for the release via a primary cooling agent leak and with respect to the helium purification plant.

A remnant in the building or a filtering of the release activity is not being considered but assumed, so that the release value in Table 5.8.1-1 with respect to the release location have been conservatively calculated.

Nuclide	Activity Release in Bq (Short-term Release)
Kr 83 m	2.1 E+10[1)]
Kr 85 m	7.6 E+10
Kr 85	1.6 E+10
Kr 87	7.8 E+10
Kr 88	1.6 E+11
Kr 89	2.8 E+10
Kr 90	1.2 E+10
Xe 131 m	2.3 E+10
Xe 133 m	5.5 E+10
Xe 133	2.3 E+12
Xe 135 m	1.8 E+10
Xe 135	2.4 E+11
Xe 137	4.8 E+10
Xe 138	9.3 E+10
Xe 139	1.5 E+10
Total noble gases	3.2 E+12
J 131	1.5 E+ 8
J 132	1.8 E+ 9
J 133	8.9 E+ 8
J 134	4.7 E+ 9
J 135	1.6 E+ 9
Total iodine	9.1 E+ 9
Sr 89	3.3 E+ 5
Sr 90	5.6 E+ 5
Ag 110 m	2.0 E+ 5
Cs 134	1.6 E+ 6
Cs 137	2.9 E+ 6
Total long-lived elements	5.6 E+ 6
Rb 88	5.3 E+10
Cs 138	2.0 E+10
Total inert gas followers	7.3 E+10
Tritium (H 3)	5.6 E+12
C 14	6.0 E+10

[1)] 2,1 E+10 = 2,1 x 10^{10}

Table 5.8.1-1: Activity release in the pressure discharge phase as a consequence of a leak in the main pipeline

Core Heating Phase

Based on our calculations, during the core heating phase, approximately 1 % of all fuel elements would reach a temperature >1500° C for a maximum 32 hours. All the remaining fuel elements are exposed to lower temperatures.

These increased fuel element temperatures lead, after the pressure discharge, to an additional activity emission via the release of fission products from the fuel elements.

The applicant determined the fission product inventory via an ORIGEN combustion calculation.

With regard to the fission product release, the applicant took into consideration the following issues:

- the fission product release out of intact fuel element particles with a subsequent diffusion through the graphite matrix,
- the fission product release out of defective fuel element particles with a subsequent diffusion through the graphite matrix,
- the release fission product from the graphite matrix sectors contaminated with uranium and silver,
- the adsorption/desorption of fission products at the graphite reflector and the fuel element graphite coat.

The damaged particles fraction is described within the calculation as an disruption temperature function. A constant defective particle rate resulting from manufacture-dependent and radiation-dependent particle decay in a temperature range of up to 1200° C is assumed in the basic calculations. Starting at 1200° C, a disruption-related particle decay is expected, which increases exponentially with a further temperature increase until 1600° C, wherein it reaches a rate of 5×10^{-4}. Both corner values are derived as follows:

- The defective particle fraction at 1,200° C results both from the maximum admissible manufacture-related defective particle fraction including the uranium-contaminated graphite matrix worth 6 x 10^{-5} and the radiation-determined fraction of damaged particles, which was derived as ten times the 95 % confidence value determined in radiation experiments on approximately 200,000 coated particles, none of which received any damage.

- The damaged particle fraction at 1,600° C is derived in the attached documentation /D 3.3-2, D 5.8-2, D 2.5.2-1, D 5.8-3/ based on the experimental exposure of 65,600 irradiated particles to a temperature of 1,600° C, which did not produce any damage. The particles were embedded in four LEU-TRISO fuel elements. Reaching respectively 3.5 %, 7.5 %, 7.6 % and 8.0 % FIMA (fissions per initial metal atom), the later had not exactly achieved the planned HTR module combustion of 80 000 MWd/Mg. Two further fuel elements containing approximately 32,800 particles all together were additionally added and heated up to 1,600° C, whereas both elements reached respectively 8.9 % and 8.6 % FIMA, which corresponds to the planned combustion discharge of the HTR module fuel elements (5.8-4). Neither of those two fuel elements suffered any damages as a result of their heat treatment. This circumstance broadens the statistical basis for defective fraction determination. However, the applicant proceeds with the assumption of a much higher fraction of damaged particles based on the four fuel elements experiment.

The applicant used the FRESCO program /D 3.3-2, B 146/ for the fission product release and transport calculation.

The fission product release out of both coated particles and the graphite matrix is described as a single-phase diffusion based on the differential Fick equation. In order to consider diverse materials, the particle is subdivided into a maximum of five zones according to its structure – nuclear fuel, protective layer, internal pyrocarbon layer, silicium carbide layer, external pyrocarbon layer. The graphite matrix represents an additional zone. The fuel elements and particles are considered as symmetrical pebbles. It is also assumed that the particles are homogenously distributed within the fuel element internal zone. The fission element transport through every single zone is described based on a diffusion equation where every zone is assigned a locally constant, effective diffusion coefficient. In the case of intact coating layers, the respective temperature-dependent diffusion coefficients of the coating materials are applied.

For the treatment of damaged fuel particles, it is assumed that defective particles without any coating are spontaneously available in the fuel element at the point of reaching the relevant temperature. The fission products that have diffused already out of the particle core into the coating are treated as directly released subjects.

The sorption of metallic fission products at the fuel element surface is taken into consideration based on the respective boundary conditions applied for the differential equation solution with respect to the diffusion process from the fuel element surface into the cooling gas. This boundary condition inference is based on the assumed occurrence of a spontaneous balance between the concentration on the fuel surface and the gas phase at any moment.

The primary circuit internal retention effects, such as the adsorption on the ceiling reflector, are considered generally with a retention factor of 10.

The filtration installation of the safeguarded low-pressure circuit is equipped with high efficiency and iodine filters and is designed to take over the outlet air filtration in the core heat-up phase. However, its availability is not redundant. With regard to the suspended matter, the outlet air filtration installation can be used as substitute. This installation does not have a iodine filter.

Therefore, the applicant presents in his safety report the core heat-up phase emissions through the 60 m high chimney both with and without functional involvement of the safeguarded low-pressure circuit.

The applicant has calculated the released primary circuit coolant according to the attached document (D 5.8-1) derived from the temperature distribution within the primary circuit and the resulting volume expansion by 9 %. The maximum core temperatures outlined in the safety report are evaluated in Chapter 5.4.1.1. Since the applicant's estimation of average primary circuit temperatures in his note is correct, we estimate that the temperatures on which the primary helium's volume expansion is based and the volume expansion resulting therefrom to be correct.

The FRESCO program has been developed for calculating the fission product release from the pellet-shaped fuel elements in KFA (B 146, B 132).

In order to verify the model, the applicant has developed a simplified computation program for the determination of the iodine release values, which is rooted in an analytical solution to the diffusion equation. A comparison between these results and the FRESCO results shows a good correspondence (D 3.3-2).

The diffusion coefficients used for the relevant nuclides, except for the iodine, are derived from the so-called HBK Standard Data Set (B 133). The HBK standard data set comprises the generally acknowledged diffusion coefficients of relevant nuclides. These were determined from extensive experimental work. An experimentally determined diffusion coefficient is also applied to iodine (B 148).

Based on these diffusion coefficients, numerous heating experiments with irradiated fuel elements and loose particles were precalculated and and cross calculated. In the case of the Cs 137 nuclide, which is emerging in radiologically significant quantities, we compared the cross calculation with the measuring results. At a temperature of 1,600° C ,the cross calculation results exceed the measuring results by several value ranges.

This is generally attributed to the diffusion coefficent values, which are set too high. The applied diffusion coefficients were derived based on particles produced back in the 1970s. Meanwhile, the particles are manufactured in a much-improved procedure and show an improved retention performance, and therefore smaller diffusion coefficients. This applies also for strontium release too.

The cross calculation of the iodine release resulting from the experimental verification of the applied iodine diffusion coefficient lay too clearly above the measured results within the relevant temperature range.

The damaged particle fraction dominates the release calculations. The damaged particle fraction used by the applicant and its statistical evaluation were checked by us on the basis of characteristic experiments and based on the relevant literature. It should be pointed out that the applied particle damage rate represents a conservative defective particle fraction in the case of a core heating disruption.

In conclusion, we note that the used FRESCO program, including the input data is feasible for determining the release of radioactive materials in the case of a core heating disruption in a conservative manner.

Since the note (D 3.3-2) was calculated only for seven long lived nuclides, for the Kr 85, Xe 135, and J 133 nuclides we derived the release rates out of core inventories and the release of calculated main nuclides.

The chemical form of iodine is elementarily assumed conservatively with 100 %.

The primary circuit activity release has been positively random checked by us via a reexamination. The release values out of a damaged coated particle are comparable with the calculation results of the Juelich Nuclear Experimental Plant.

Due to the design of the non-redundant safeguarded low-pressure circuit filtration plant, we assume that the plant might no longer be available, and that the halogens therefore has failed. The solid matter filtration is assumed with an efficiency grade of 99.9%, since the outlet air filtration unit can be described as a redundancy to the secured low-pressure circuit and is also equiped with a high-efficiency filter, but without a iodine filter layer.

The release values resulting from these assumptions are presented in Table 5.8.1-2.

The release rate of radioactive substances into the environment grows slowly with respect to the time span of the core heat-up phase, it reaches it's peak, and then drops again slowly in correspondence with the core release rate, whereas the release rate of the primary helium decreases slowly and constantly in the core heat-up phase relevant time span starting 10 hours after the occurance of an disruption. Therefore, alternative release intervals should be taken into consideretaion for the purpose of a deterministic disruption expansion calculation. For this reason, the applicant has chosen the time intervals in such a way that the maximum release within the core heat-up phase during a period of eight hours corresponds to an 8-hour interval. The respective time-related staggering is also presented in Table 5.8.1-2.

If iodine filtration through the safeguarded low-pressure circuit is assumed, the iodine release rates would reduce by two value ranges.

Activity Release in Bq within the Respective Time Interval

Nuclide		0 – 8 h	8 – 24 h	24 - 27 h	>72 h	Total
Kr	85	1.1 E+6	9.0 E+7	8.9 E+8	5.4 E+8	1.5 E+9
Xe	133	2.6 E+8	1.9 E+10	1.6 E+8	8.5 E+10	2.7 E+11
Xe	135	3.0 E+7	9.9 E+8	1.4 E+9	8.3 E+7	2.5 E+9
J	131	1.3 E+7	92 E+8	8.1 E+9	4.1 E+9	1.3 E+10
J	133	2.5 E+7	1.2 E+9	4.6 E+9	1.0 E+9	6.9 E+9
Sr	89	3.0 E+1	5.5 E+3	1.1 E+5	1.1 E+5	2.2 E+5
Sr	90	1.4 E+0	2.6 E+2	5.1 E+3	5.0 E+3	1.0 E+4
Ag	110m	1.4 E+2	5.8 E+3	4.0 E+5	8.4 E+5	1.2 E+6
Cs	134	1.0 E+3	5.1 E+4	8.1 E+5	8.7 E+5	1.7 E+6
Cs	137	1.2 E+3	5.9 E+4	9.4 E+5	1.0 E+6	2.0 E+6

Alternative Time Intervals

Activity Release in Bq within the Respective Time Interval

Nuclide	0–34 h	34–42 h	42-58 h	58–106 h	>106 h	Total
Kr 85	2.4 E+8	1.5 E+8	3.1 E+8	6.4 E+8	2.1 E+8	1.5 E+9
Xe 133	5.8 E+10	3.4 E+10	5.8 E+10	9.7 E+10	2.7E+10	2.7E+11
Xe 135	1.8 E+9	3.6 E+8	2.8 E+8	7.7 E+7	2.5 E+6	2.5E+9
J 131	2.7 E+9	1.6 E+9	2.8 E+9	4.6 E+9	1.3 E+9	1.3E+10
J 133	4.0 E+9	1.1 E+9	4.9 E+8	1.2 E+9	1.5 E+8	6.9E+9
Sr 89	2.1 E+4	1.9 E+4	4.1 E+4	1.0 E+5	3.7 E+4	2.2E+5
Sr 90	1.0 E+3	8.8 E+2	2.0 E+3	4.7 E+3	1.8 E+3	1.0E+4
Ag 110m	3.6 E+4	3.2 E+4	1.6 E+5	6.7 E+5	3.4 E+5	1.2E+6
Cs 134	1.7 E+5	1.4 E+5	3.1 E+5	7.8 E+5	3.3 E+5	1.7E+6
Cs 137	2.0 E+5	1.6 E+5	3.6 E+5	9.1 E+5	3.8 E+5	2.0E+6

Table 5.8.1-2: Activity release in the core heat-up phase after a break in a main pipeline (DN 65)

5.8.1.3 Leakage of a Primary Cooling Agent Measuring Pipe

A break in a measuring pipe with an internal diameter of </= 10 mm is assumed. The resulting reactor pressure relief occurs slowly. The control switch is correspondingly delayed and is assumed at 50-bar pressure. The pressure balance with ambient at the leakage point is only achieved several hours later.

The helium outflow takes place at sonic speed. As a result, even a break in a pipe with a 10mm nominal width requires several hours. The control switchoff at 50 bar expected by the applicant should be considered as pessimistic with reference to the criterion "negatively sliding primary pressure limit > app. 180 mbar/min." According to these assumptions, the quick switching would occur after nearly one hour. The absence of the previously mentioned stimulus criterion presupposes smaller break cross sections than those of the nomal width of 10.

It is unnecessary to assume a quick switching at a later point, because it might be assumed that tiny leaks could be discovered within an hour, based both on the primary circuit pressure loss and the reactor activity increase.

Activity Release

Until a pressure balance with the leakage surroundings is reached, the entire primary circuit inventory as well as the radioactive materials desorbed during the pressure drop from the surfaces into the cooling agent are released. No dust discharge occurs through the leakage point.

In our opinion, the complete pressure relief caused by the break of a measuring pipe represents the discussed case with respect to the radioactive substance release. Since the helium flow velocity is only increased to a considerable extent within the measuring pipe, a dust discharge should not be taken into consideration. The primary circuit activity release outlined in the applicant's note (D 3.3-2) is correct with respect to the radioactive decay because it takes into account the radioactive quantities released by the break of a main pipe (DN 65) via the surface desorption and as primary cooling agent inventory. According to the basis for the disruption calculation (B 14), we assume that the function of a filter unit is available on demand, starting 10 minutes after the disruption occurrence. We accept the filtration efficiency degrees applied by the applicant, except the one for the iodine filter. We do not take the latter into consideration just like the core heat-up approach due to missing redundancy.

With respect to the release mechanisms, we assume that part of the released iodine is transformed into an organic iodine substances. We assume that this fraction with 50 % and the residue should remain available in elementary form. In this way, the radioactive substances presented in Table 5.8.1-3 are released through the chimney. A conservative assumption is made, that the release takes place within the first eight hours after the disruption occurrence, although smaller leaks might prolong the emission.

We have calculated the fraction of unfiltered solid matters released through the chimney within the first 10 minutes after the accident occurrence.

We have esimated conservatively an air exchange of 5/h at the leakage point and a primary circuit leakage rate of 1 %/min. Herein 3.2 % of the radioactive solid matter inventory contained in the primary circuit helium are released unfiltered within the first 10 minutes after the disruption occurrence. The respective nuclide vector is shown in Table 5.8.1-3.

Should iodine filtering through the secured low-pressure circuit be assumed, then the iodine release rates would be decreased by two orders of magnitude.

Nuclide	Activity Release in Bq	
Kr 83 m	1.2 E+10	
Kr 85 m	4.6 E+10	
Kr 85	3.2 E+ 8	
Kr 87	4.0 E+10	
Kr 88	9.7 E+10	
Kr 89	5.9 E+ 9	
Kr 90	2.2 E+ 9	
Xe 131 m	1.2 E+ 9	
Xe 133 m	1.1 E+10	
Xe 133	2.3 E+11	
Xe 135 m	4.6 E+ 9	
Xe 135	1.3 E+11	
Xe 137	1.0 E+10	
Xe 138	2.7 E+10	
Xe 139	2.8 E+ 9	
J 131 el	6.5 E+7	
J 132 el	3.4 E+8	
J 133 el	3.9 E+8	
J 134 el	4.2 E+8	
J 135 el	5.0 E+8	
J 131 org.	6.5 E+7	
J 132 org.	3.4 E+8	
J 133 org.	3.9 E+8	
J 134 org.	4.2 E+8	
J 135 org.	5.0 E+8	
Sr 89	3.0 E+03	of which 3.2 E+2 filtered and 2.7 E+3 unfiltered
Sr 90	8.2 E+01	of which 8.7 E+0 filtered and 7.3 E+1 unfiltered
Ag 110 m	6.9 E+02	of which 7.7 E+1 filtered and 6.1 E+2 unfiltered
Cs 134	9.4 E+03	of which 1.0 E+3 filtered and 8.4 E+3 unfiltered
Cs 137	1.9 E+04	of which 2.2 E+3 filtered and 1.7 E+4 unfiltered
Rb 88	4.2 E+08	of which 4.6 E+6 filtered and 4.2 E+08 unfiltered
Cs 138	1.5 E+08	of which 2.5 E+6 filtered and 1.5 E+08 unfiltered

Table 5.8.1-3: Activity release into the environment after a break in a measuring tube

5.8.1.4 Failure of the Heating Pipe of the Steam Generator with Long-term Breakdown of the Water Elimination and Primary Circuit Pressure Regulation

A break in a 2F heating pipe is assumed. Thereafter, until the activation of the reactor protection actions (moisture measurement in the primary circuit), 54 Kg of water/steam convert in the primary circuit. During the steam generation pressure relief process with the purpose of a pressure comepensation with the primary circuit, another 117 Kg of water/steam are transferred. After this, the remaining portion in the steam generator amounts to 300 Kg of water/steam. A conservative assumption is made, that a total of 600 Kg of water/steam pass into the primary circuit.

A detailed disruption scenario is described and evaluated in Chapter 5.4.3.

Activity Release

The water or steam entering the primary circuit corrodes the fuel element surfaces. Water gas is produced, which causes a pressure increase in the primary circuit. The safety valves included in the primary circuit pressure relief system are activated at 69 bar after approximately 3.5 hours (see Chapter 5.4.3) and close after approximately 30 minutes at 62 bar. On the occasion of the aforementioned pressure relief, approximately 10 % of the radioactive substances carried into the primary circuit by the heating pipe leakage are released into the reactor building. Released inventory reaches the chimney through the filter unit.

Except for the primary gas activity in the beginning of a disruption, three additional activity sources occur in the steam generator surfaces: a re-mobilization of the steam generator surfaces, a mobilization by the fuel element corrosion, and an iodine and noble gas release by UO_2 oxidation.

- Remobilization of the Steam Generator Surface

A mechanical decontamination is produced for the surface, which was affected by the water jet or steam jet. An area of 12 m^2 with a decontamination of 100 % is assumed. This corresponds to nearly 6 % of the steam generator total area. However, steam generator surface sediments are not assumed as equally distributed but evenly exponentially decreasing over the entire area. In contrast, the iodine is considered as being 100 % evenly distributed over the preheater and floor calotte.

The steam generator fraction of the radioactive substance inventory within the primary circuit is estimated at 95 % for Sr and Cs, 90 % for Ag, and 80 % for J. Of these, fractions of up to 5 % are assumed to be dust-related and 80 % of the dust-unrelated fraction are considered to be related to the adsorptiion (Cs, Sr) and therefore remobilizable. For J, this fraction can even reach 100 %. In the case of Ag, this assumption is not relevant , since Ag is chemically inert. From these assumptions result then the activity release into the primary circuit through remobilization caused by a water/steam jet for the different leakage locations. Except for iodine leakages in the overheater, a maximum release as in the case of every leakage spot maximum surface activity was assumed.

For the description of the remobilization by the reaction with water and with steam interaction, the applicant rests upon the wash-off experiments of the Juelich nuclear experimental plant (B 144).

According to these, the initial Cs reaction fractions on the Incoloy 800 surfaces reach 6×10^{-2} min^{-1} for water and 5×10^{-6} min^{-1} for steam. A water overflow should be expected in the preheater sector.