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Safety Assessment of the Design of the Modular HTR-2 Nuclear Power Plant

- Summary -

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1 Introduction

1.1 Extent and Course of the Task

In April 1987 the Siemens and Interatom companies applied for a provisional decision according to § 7a of the German Nuclear Energy Act concerning the design of a modular HTR-2 Nuclear Power Plant for combined generation of electrical power and process steam or heat for district heating, respectively. The application was submitted to the Ministry for the Environment of the German state of Lower Saxony.

TÜV Hannover/Sachsen-Anhalt e.V. (at that time TÜV Hannover e.V.) was contracted by the aforementioned ministry to assess the safety of the modular HTR-2 NPP in the licensing procedure. TÜV Rheinland e.V. was subcontracted for specific tasks by TÜV Hannover/Sachsen-Anhalt e.V..

The basis of our first preliminary investigations was the applicants' safety analysis report dating from April 1987, which they had submitted to the licensing authorities as addendum to the application. During a sequence of specialists' meetings taking place in the second half of the year 1987 and in the first half of the year 1988 we substantiated our request for further, more detailed reports concerning the design of the modular HTR-2 NPP and we proposed several modifications of the safety concept to be carried out by the applicants.

As a result of our proposals and requests the applicants modified the HTR-2 design and submitted further technical reports for our safety assessment. Additionally, the safety analysis report /1/ was revised. The task to update and complete the documents was finished early in 1989.

In April 1989 the application for a previsional decision was withdrawn; the licensing procedure was terminated by the Ministry of the Environment and the contract with TÜV Hannover/Sachsen-Anhalt e.V. was cancelled.

At that time the task to assess the safety of the modular HTR-2 NPP was in an incomplete, but largely advanced state. For this reason TÜV Hannover/Sachsen-Anhalt e.V. offered to the Federal Ministry for Research and Technology to perform an assessment of the HTR-2 NPP safety concept - in this case not related to a licensing procedure - and to evaluate, if this concept complied with the general safety requirements on nuclear installations in Germany and which conceptual features should be modified. This task aimed at ensuring that possible future research and development activities were compatible with the German safety standards concerning nuclear facilities.

TÜV Hannover/Sachsen-Anhalt e.V. was entrusted with this task and submitted its safety assessment report /2/ of roughly 900 pages in October 1989. In the following we summa-

rize the relevant results of the aforementioned report and we demonstrate by several examples, which modifications of the NPP design methodology resulted from the safety assessment. *(Remark for comprehension: Although the tasks for the Lower Saxony Ministry of the Environment and the Federal Ministry for Research and Technology differ markedly as the second task does not fall within the scope of a licensing procedure, we consistenly use the term "applicants" for the Siemens and Interatom companies in this report to avoid possible misunderstandings.)*

1.2 Assessment Criteria and Design Basis Requirements

The German Nuclear Energy Act /3/ and the Radiological Protection Ordinance /4/ form a general legal basis for planning, construction, operation, and decommissioning as well as supply and waste management of nuclear facilities and thus define the framework for safety assessment of these installations. They are not restricted to specific plant concepts and technical construction details. The requirements imposed by the Nuclear Energy Act and the Radiological Protection Ordinance have to be met under any circumstances.

The legal basis is supplemented by a series of technical rules and guidelines, which largely relate to technical concepts and which had been elaborated continuously accompanying progress of nuclear technology. They aim at defining the state of the art and shall enable the manufacturer of a nuclear plant as well as the expert to apply uniform evaluation criteria in their tasks. Most of the rules and guidelines apply to LWRs. Thus the German rules and guidelines in nuclear technology mainly relate to the physical and technological features of LWRs and especially to those of pressurized water reactors, which are characterized by a nuclear core design with high power density. For this reason the rules and guidelines cannot be transferred to reactor concepts differing substantially from the LWR without modification.

Gas-cooled high temperature reactors are characterized by pronounced features of inherent safety against reactivity transients and disturbed heat removal. This applies especially to units with low power and low power density in the reactor core. Thus it would have been inadequate to transfer the unmodified requirements on LWRs to the modular HTR-2 NPP, particularly as in the latter case a reactor concept had been developed by consequent exploitation of low-power HTR safety characteristics, which even in the case of failure of all active cooling systems and in the loss-of-coolant accident will limit the fuel temperature in such a way that no relevant release of radioactive fission products will occur. This property reaches far beyond technologically realized standards, which implies that even the rudimentarily existing German rules and guidelines for HTRs had to be adapted before applying them to the modular HTR-2 NPP.

Therefore we applied in our assessment of the HTR-2 safety concept - apart from the concept-independent legal basis - the technical rules and guidelines only to such an extent as they are compatible with the concept itself. In our report to the Federal Ministry for Research and Technology we have justified in detail our procedure to deviate from technical rules and guidelines, when necessary. It was, however, beyond the scope of our task to perform a complete assessment of the applicability of all rules and guidelines in nuclear technology to the modular HTR-2 NPP design.

According to the Nuclear Energy Act the protective measures against damage from nuclear energy have to comply with the state of science and technology. This means for a prototype plant that additionally to the modified technical rules and guidelines the current state of research and development must be taken into consideration. For this reason we referred extensively to results presented in publications.

1.3 Assumed Site Characteristics

The conceptual design of the modular HTR-2 NPP was developed without referring to a concrete site of the planned NPP. For this reason site characteristics were assumed in such a way that they are representative for a large number of potential sites. Site-independent characteristics, referring e.g. to aircraft impact, were assumed according to the applicable technical rules and guidelines.

In our safety assessment we verified the site characteristics given by the applicants in their safety analysis report only with respect to completeness and conclusiveness. Further we verified if the chosen procedures to generate the site characteristics complied with the applicable German rules. We mainly paid attention to those characteristics, which could affect conceptual design and operation of the plant according to our general experience with nuclear facilities.

The result of our evaluation was that the assumed site characteristics, given by the applicants in their safety analysis report, take into account all relevant loads and design requirements to be imposed on nuclear facilities site-independently and cover to a large extent the properties of potential sites. As, on the other hand, the assumptions cannot replace all realistic characteristics, it will be necessary to evaluate a concrete site for the modular HTR-2 NPP for its suitability and compatibility with plant design. Therefore we have to state that the results of our safety assessment are valid only for those sites, which are compatible to the assumed site properties. As an example we based our evaluation of the emergency power supply concept on reliability data, which are typical for Germany.

2 The Modular HTR-2 NPP

2.1 Basic Safety Design Features

The nuclear core design of the modular HTR-2 NPP is such that even in the case of an assumed long-term failure of all active installations for heat removal and a simultaneous failure of the scram system the fuel temperature will not exceed its acceptable maximum value (fuel design temperature), which amounts to 1620 °C. This inherent safety feature is mainly caused by the large difference between fuel temperature under normal operation conditions and fuel design temperature, which will cause an intrinsic shut-down of the reactor, as well as the low power density of the reactor core in conjunction with its slim geometry and the design of the surrounding structural components, which will enable passive heat removal from the core.

The aforementioned scenario is a hypothetic combination of two different incidents and extends beyond design basis assumptions. It demonstrates, however, the inherent safety features of the modular HTR-2 NPP, which form the basis of its safety concept and which have to be taken into account in our safety assessment. In the following we will deal with the question, how the dominant protective requirements to grant the necessary safety of nuclear power plants - i.e. the required shut-down safety margin, decay heat removal and retention of radioactive substances - are met by the modular HTR-2 NPP, which is characterized by its inherent safety features.

Shut-down Safety

Due to its design the reactor core can be shut down already by insertion of neutron absorbers in vertical openings of the side reflector. There are two different measures, which are called the "first" and "second shut-down system".

The first shut-down system consists of the reflector rod system. According to the safety analysis report this system is designed in such a way that it will shut down the reactor sufficiently fast and keep it in the state "hot, below criticality" both from normal operation conditions and in design basis accidents. The system design takes into account the assumed failure of the most reactivity-efficient reflector rod. For thermal decoupling of the reactor and the steam generator as a prerequisite to keep the reactor in the state "hot, below criticality" the primary circuit blower must be shut off, too, i.e. both the first shut-down system and the primary circuit blower are shut down automatically and simultaneously by the reactor protection system. Due to its task to keep the reactor in the state "hot, below criticality" the first shut-down system is also called "hot shut-down system".

The second shut-down system consists of the small-sphere shut-down system. This system is designed such that it will shut down the reactor from normal operation in those

cases, which do not require fast reactivity changes, and keep it continuously below criticality at 50°C, the lowest operation temperature. The small-sphere shut-down system is also called "cold shut-down system". This system will be initiated by manual action and not automatically.

According to the aforementioned inherent safety properties of the reactor, it can also be shut down by interrupting the primary coolant flow, which means the primary coolant blower is turned off. This action would not cause an immediate interruption of heat generation; however, as a result of disturbed heat transfer to the secondary circuit the core temperature will rise and the reactor will stabilize below criticality on a higher temperature level due to the negative temperature coefficient of the reactivity. The applicants did not take into account this effect in their shut-down concept, as shut-down of the blower and insertion of the reflector rods are always initiated simultaneously.

Decay Heat Removal

Due to the intrinsic property of the modular HTR-2 reactor to remove the decay heat after shut-down passively by heating up the surrounding structural components, there is no need to integrate the coolant circuits, especially the secondary circuit, into the safety concept, as the fuel temperature will be below fuel design temperature under any circumstances. Thus the main protective task of the active coolant systems is to limit the temperature of the concrete structures surrounding the reactor core and of the reactor pressure vessel including its components to acceptable values.

This protective task is valid for the concrete structures also under normal operation conditions, as the heat due to energy losses of the reactor pressure vessel will heat up the reactor cavity. For this reason the reactor cavity is equipped with a surface cooler. The supporting components of the reactor pressure vessel and the socket integrated into the vessel bottom to join the fuel removal pipe are cooled separately. If heat transfer to the main heat removal system is interrupted, the surface cooler will be active to prevent excess temperatures due to decay heat after shut-down of the reactor.

The applicants have performed analyses to demonstrate that there is no need for short-term availability of the active cooling systems, as the design temperatures of the reactor cavity concrete structures will only be exceeded after 15 hours.

Retention of Radioactive Substances

The spherical fuel elements of the modular HTR-2 NPP contain the fuel in multiply coated particles (TRISO-particles), which are embedded into a graphite matrix. Due to

these features the fission products - which form the bulk of all radionuclides in the reactor core - will be enclosed nearly completely, provided that the fuel design temperature is not exceeded. Thus the intrinsic limitation of the fuel temperature and of mechanical loads to values below design limits due to its inherent safety features is one of the basic properties of the modular HTR-2 NPP.

The radioactive substances released to the coolant originate either from only few defective coated particles or from fission or activation of uranium traces in the graphite matrix of the fuel elements. It has been demonstrated successfully both by experiments and by operation experience that under the given circumstances the resulting coolant activity in the primary circuit will be very low.

In a loss-of-coolant accident the fission gas activity of the coolant and part of the plate-out activity on the primary circuit surfaces would be released to the reactor building and via the ventilation stack to the environment. According to different analyses the resulting radiation exposure in the environment would be far below the accident dose limits laid down in the German Radiological Protection Ordinance. For this reason the design of the modular HTR-2 NPP does not include a gastight containment.

2.2 Technical Design

In this section we give exemplary results of our safety assessment, which refer to important components and systems of the modular HTR-2 NPP. For our summary we have selected those systems and components as well as those results, which are relevant for the design of the modular HTR-2 NPP and its operational and accident behaviour. The complete and consistent results of our investigations are compiled in our safety assessment report of the modular HTR-2 reactor design.

2.2.1 Nuclear Steam Generation System

The modular HTR-2 NPP consists of two reactors, each of which being designed for a thermal power of 200 MW. For this reason two independent nuclear steam generation systems are foreseen, which to a certain extent are connected to common auxiliary systems.

Each steam generation system has the task to transfer the heat generated by fission in the reactor core to the feedwater-steam circuit in the steam generator. The coolant in the primary system is highly purified helium. Under normal operation conditions the heat transport occurs by forced convection, where the coolant flow is maintained by the primary circuit blower. If the blower is turned off, the decay heat will be removed from the

core by radiation and natural convection via the reactor pressure vessel to external cooling systems (surface cooler).

The nuclear steam generation system consists of the following main components:

- reactor pressure vessel (RPV) including core, core components and shut-down systems,
- connecting pressure vessel including the hot gas duct,
- steam generator pressure vessel including its components, the primary circuit blower and the blower isolation valve.

These three main components form a common pressure vessel unit. The pipes to adjoining systems are connected to sockets and are led out of the primary cell by means of wall passages. All valves in these pipes are arranged outside the primary cell. This applies to the primary circuit isolation valves, too. The steam generator pressure vessel is shifted to a lower level with reference to the reactor pressure vessel and arranged at the side of the latter. Thus the hot gas duct will be very short. Due to the low positioning of the heat sink as compared to the heat source, any interruption of forced convection will be followed by only poor natural convection. This complies with the intention that the decay heat shall only be removed from the RPV walls to the surface cooler.

the effects of incidents caused by failures in the secondary circuit, especially water ingress into the primary circuit, are limited distinctively by arrangement and design of the primary circuit components. The feedwater and steam pipings are arranged on a comparatively low level and decoupled from the RPV. Thus a direct contact of the reactor core with water in its liquid state can be excluded for all relevant design basis accidents; only a contact to a helium/steam mixture has to be taken into consideration. All investigations concerning fuel element and material corrosion can be based on this boundary condition.

The arrangement of the primary circuit components in the modular HTR-2 NPP is advantageous for operational reasons and especially for safety aspects. On the other hand the primary circuit design as a basis-safe pressure vessel unit means high requirements on design, construction, materials, weldings and the support concept of the components.

Fuel Elements

The reactor core consists of about 360,000 spherical fuel elements, which are arranged within the ceramic and metallic core components in the form of a pebble bed. The spheres have an outer diameter of 60 mm. The fuel is embedded within an inner zone of the spheres (50 mm ∅) and enclosed in particles consisting of the fuel-containing kernels

(0.5 mm ∅), which are surrounded by several pyrolytically deposited carbon layers and a layer of silicon carbide (SiC). These coated fuel particles are distributed evenly within a graphite matrix.

The task of the spherical fuel elements is to generate heat by fission of the nuclear fuel. Loads resulting from handling, operation and possible accidents and affecting both the fuel elements and the adjoining components have to be limited in such a way that the dominant protective requirements can be met. For this reason the fuel elements have to be designed such that the

- fission gas release will be limited to acceptable values,
- strength of the spherical fuel elements will be sufficient,
- fuel element corrosion will range within acceptable limits,
- dimensional stability will be sufficient.

In their documents the applicants have described the planned fabrication methods of the fuel elements for the modular HTR-2 NPP and indicated, which quality assurance methods are foreseen to grant an even and sufficiently high quality level in fuel manufacture.

The existing manufacture and operation data concerning the foreseen fuel type are mainly based on the experience made with the AVR reactor. Additional extensive experience with different fabrication steps stems from the manufacture of THTR reactor fuel elements. Furthermore fuel elements and fuel samples were manufactured for a great number of irradiation experiments in material test reactors.

Ensuring the product properties - especially fission product retention - is of conceptual importance in safety assessment of the modular HTR-2 NPP. According to our opinion the applicants have demonstrated successfully by referring to extensive manufacturing experience made with fuel elements and fuel samples for irradiation experiments and the AVR reactor that fuel elements with properties required for the modular HTR-2 NPP can be manufactured. We further expect that the quality assurance system and the correspondig manufacture and inspection methods are suitable for transition to a large production scale and thus ensure the required fuel properties.

For our results concerning fission gas release from the fuel please refer to section 4.2.

Core Components

The relevant construction feature of the ceramical core components consisting of either graphite or carbon material is to divide the structure into single blocks, as has been demonstrated succesfully in the high temperature reactors AVR and THTR. The metallic core

core components form a support structure and are designed to bear the loads due to normal operation and under accident conditions. The ceramic core components, which form the upper, side and bottom reflectors, are made from graphite materials and will due to their neutron-physical properties reflect the neutrons escaping from the core to the pebble bed again. The outer part of the reflector will be made from carbon material, which has a lower heat conductivity as compared to graphite and thus will protect the adjoining metallic components against excessive temperature loads. The bottom layer and the neighbouring layer of the side reflector contain boron as a neutron absorber to reduce the neutron irradiation of the reactor pressure vessel, the metallic core components and the hot gas duct.

In our safety assessment we have verified by performing design calculations that thermally induced displacements of the core structure components will not affect shape and dimensions of the core enclosure and of the openings for the control rods and shut-down eqipment as well as for the coolant gas flow.

Shut-down Systems

In section 2.1 we have already described the shut-down concept. In the following we will summarize the results of our evaluation concerning both shut-down systems of the modular HTR-2 NPP with respect to their construction and functional safety.

According to the shut-down concept of the modular HTR-2 NPP the reflector rod system is the first shut-down system and serves - in conjunction with the simultaneous blower shut-down - as a scram system, which is designed to reduce the reactor power from normal power operation and under design basis accident conditions and to keep it in the state "hot, below criticality" as long as necessary, even if the most efficient reflector rod has failed.

Apart from this safety-related task the reflector rods are designed for reactivity control of the reactor under power operation conditions.

The reflector rod system corresponds in its relevant properties to that of the THTR reactor. It consists of six reflector rods and their drive and control units. The reflector rods are suspended by connecting them to their drives by chains and are moved up and down in the openings of the side reflector. The six control rod drives are arranged above the the so-called thermal upper shielding within the reactor pressure vessel.

The rod insertion limitation shall ensure the shut-down reactivity required to scram the reactor by the reflector rods at any time and from all operational modes.

The strength design of the reflector rod components and of their drives must be compatible with the loads to be expected, and their operational behaviour has to be verified experimentally and by evaluation of operation experience made with comparable systems to ensure an undisturbed and highly reliable shut-down behaviour of the reflector rods, even after long-term operation.

For this purpose the applicants have planned an evaluation of already performed prototype experiments with THTR-specific components as well as experiments to be carried out with a prototype specific for the modular HTR-2 NPP. In our view the foreseen experiments, inspections and calculations will be suitable to demonstrate that the detailed design and construction will meet all requirements of the KTA-rules to be applied to the first shut-down system.

The initiation concept of the scram system has been realized in comparable nuclear power plants and has proved to be reliable.

The small-sphere shut-down system is the second shut-down system and shall shut down the reactor from normal operation in those cases, which do not require fast reactivity changes, and keep it continuously below criticality at 50°C, the lowest operation temperature. In case of a scram it will in conjunction with the reflector rod system keep the reactor in the long-term state "cold, below criticality". Apart from this safety-related task the small-sphere shut-down system is involved in reactivity control of certain normal operation modes.

The small-sphere shut-down system contains 18 units, which are independent from each other. One unit consists of the following main components:

- Storage container with container lock and cyclone separator,
- removal container,
- transportation pipe and transportation gas return pipe,
- small-sphere shut-down elements.

Shut-down of the reactor is achieved by dropping the neutron-absorbing small-sphere elements from the storage containers into openings (oblong holes) in the side reflector.

The design of the small-sphere shut-down system planned by the applicants is new. According to the state of science and technology the functional safety of shut-down installations, which have a safety-related task, has to be demonstrated. The relevant basis for such a demonstration will be a suitability test carried out with a prototype unit of the newly developed shut-down system, if the applicant cannot refer to positive operation experience.

The safety function of the small-sphere shut-down system requires to open reliably the trail for the spheres to drop from the storage containers into the oblong holes in the side reflector. Opening the trail is achieved by opening the container lock, which is designed in a very simple way and consists of only few movable parts. The relevant requirements on the function of the container lock are:

- Reliable opening,
- suppression of arching during sphere discharge,
- discharge of partial lots without damage to the spheres due to closing the lock.

We expect that these requirements can be met by the design planned by the applicants and that the necessary functional safety of the container lock can be demonstrated successfully by the foreseen suitability tests.

As the small-sphere shut-down system is safety-relevant apart from its importance in normal operation, a level limitation of the sphere columns in the oblong openings in the side reflector, correlated to the level indication of the storage containers, is required to ensure the necessary shut-down reactivity.

The small-sphere shut-down system shall be initiated by manual operation. This design feature complies with the results of the accident analysis.

Pressure Vessel Unit

The pressurized walls of the pressure vessel unit will be made from the heat-resistant, fine-grained and heat-treatable steel 20 MnMoNi 55. This material has been qualified for operation in nuclear facilities at temperatures up to 375 °C and is suitable for the planned application range.

Nevertheless, additional investigations are required to gain material data relevant for material embrittlement due to neutron irradiation in the temperature range of about 250°C, the foreseen operation temperature of the HTR-2 reactor pressure vessel. These investigations shall be performed within an irradiation program preceding operation and simulating long operation times. The aim of this program is to gain a relation between the relevant material data and neutron fluence at temperatures typical for the HTR. Additionally the applicants are planning to compare irradiation results at low and high neutron flux densities to demonstrate that simulating long operation times by high neutron fluence densities will lead to conservative results. The applicants will carry out material tests of the samples after irradiation to verify the shift of the brittle fracture transition temperature, upon which the fracture-mechanical design analyses have been based. If a higher value results as compared to design calculations, further fracture-mechanical samples will be

irradiated. To verify the progress of embrittlement due to neutron irradiation, material samples stemming from the original material of the reactor pressure vessel will be irradiated during operation of the modular HTR-2 NPP. As for plant-specific reasons these irradiation investigations cannot be carried out applying a lead factor, the sample number and the discharge frequency will be increased as compared to the requirements of the KTA-rules to predict reliably the expected increase of irradiation embrittlement between two discharge steps. The planned positions of the sample holders close to the reactor pressure vessel wall will ensure that temperatures and fluences at the sample locations are representative for the vessel material.

The pressure vessel unit is designed for the assumed failure of the largest pipe (65 mm ∅) joined to the vessel, i.e. the occurence of larger leaks is not postulated. This assumption determining the design of the modular HTR-2 NPP is justified by the applicants by referring to the planned design and surveillance measures that will exclude a failure of the pressure vessel unit during the lifetime of the plant. In their opinion especially the

- high ductility of the material,
- stress limitation by meeting the requirements of the KTA-rules,
- monitoring of possible flaws in the components after manufacture and after each operation cycle

justify the exclusion of fractures penetrating the vessel walls. Thus they exclude leakages for any wall area of the pressure vessel unit. Additionally the applicants carried out fracture-mechanical investigations of the pressure vessel unit to justify the exclusion of fractures.

According to our assessment the applied methods to justify the exclusion of cracks are suitable. We have verified the results of the applicants' investigations by performing independent calculations with our computer codes. In our opinion the applicants' reasons to exclude fractures of the pressure vessel unit are well-founded. This result is supported by the fact that the area of fracture, which can be determined by assuming a very long duration of operation extending the lifetime of the plant, but can be excluded to occur during its lifetime, will be below the area of fracture represented by the pipe with a diameter of 65 mm joined to the vessel. The fracture length occurring on the basis of these assumptions will be far below the critical fracture length, i.e. critical failure can be excluded.

Based on our evaluation of component experiments, which aimed at demonstrating safe operational behaviour of the welding between ferritic and Incoloy materials at operation temperature of 530 °C, and our crack propagation calculations we came to the result that for the designed socket of the steam duct located at the steam generator we cannot confirm the exclusion of fractures.

Therefore the applicants have redesigned the socket of the steam duct avoiding consequently weldings between ferritic and Incoloy materials in this area characterized by design temperatures above 350 °C.

The measures taken in redesigning are suitable to minimize the loads on the components of the socket, especially by avoiding stress effects due to inhibited thermal expansion of the different materials at temperatures above 350 °C. The stress calculations performed by the applicants for the most important operational modes covering the loads to be expected meet the requirements to be imposed on fracture mechanical calculations. We have verified these caculations by performing independent calculations with our computer codes and found that for the design lifetime of the plant the occurence of leakages and fractures can be excluded, as even the propagation of large initiating flaws will be reduced both in depth and longitudinal directions to such an extent that no wall-penetrating fracture will occur.

Further investigations on fracture propagation demonstrate that only after operation intervals extending by far the design lifetime of the plant the occurence of a leakage becomes possible. However, this will not induce a rupture of the socket of the steam duct. The possible maximum leakage cross-sections - not to be expected during the lifetime of the plant - has been determined in a sufficiently conservative manner and is covered with respect to its consequences by the design leakage cross-section. The design leakage between the primary and secondary circuit is the double-sided rupture of a steam generator heating tube. According to our safety assessment this assumption is justifiable, taking into account design, materials and planned quality assurance measures during fabrication.

In our opinion a failure of the steam generator heating tubes in the course of an assumed rupture of the feedwater or steam ducts has not to be postulated, as the heating tubes are designed for the loads from these events. We further agree with the applicants that the double-sided rupture of a heating tube has not to be postulated to occur simultaneously to the rupture of the steam duct, as only reduced pre-damaging of the HTR heating tubes as compared to the LWR will occur due to the effect of the precautionary measures in water chemistry; further the foreseen moisture measurement equipment will be able to identify very small leakages already. However, in contrast to the applicants we see the need for regular operational non-destructive tests to verify the state of the heating tubes during operation.

The design to support the pressure vessel unit provides both for bearing the operational and accident loads and for facilitating uninhibited heat extension. The support components will be designed such that on the one hand the pressure vessel unit will keep its

position in the building under the influence of outer forces and on the other hand they will not inhibit radial and axial displacements due to temperature expansion.

In our view the planned design to support the pressure vessel unit is suitable to bear all loads in horizontal and vertical direction resulting from the weight of the vessel unit as well as from operation and accidents. It further facilitates a mostly uninhibited extension of the pressure vessel unit due to temperature influence. For this purpose the applicants will provide the plant with proven and reliable sliding bearings, which are characterized by a careful finish of the sliding surfaces and well-adapted selection of the lubricant, thus avoiding an unacceptable restraint. Further the planned arrangement of guide units on the different support levels will facilitate the vessels to displace as required and to avoid unacceptable compulsive forces in those areas, where they could become relevant.

We have assessed the need to regularly inspect the support components during operation referring to their safety relevance. In our view regular operational visual inspections combined with surface inspections and gauging will be necessary, whereas the applicants plan to inspect the shock absorbers only.

It is the aim of the visual inspections to verify the unrestrained moveability of the support components and the shock absorbers as well as the absence of any deformation, damage or corrosion effect. Gauging will be applied to verify the adjusted cold and warm clearances to avoid unacceptable restraints in displacing the pressure vessel unit during operation. Surface inspections, as dye penetrant and magnetic particle tests, will be applied, if surface damage is to be assumed.

2.2.2 Secured Intermediate Cooling System Including the Surface Cooler

Exclusively the following coolant systems of the modular HTR-2 NPP are designed for safety-related tasks:

- the secured intermediate cooling system including the surface cooler and
- the secured service cooling water system.

Both systems together form a two-train redundant sequence of coolant systems. The secured coolant system is connected to the

- surface cooler,
- support of the pressure vessel unit,
- socket in the reactor vessel bottom (fuel element discharge pipe),

which shall remove the radiation and convection heat originating from the unisolated part of the reactor pressure vessel, the support of the pressure vessel unit and the fuel element discharge pipe to protect the concrete structure against excess temperatures.

Furthermore the surface cooler shall maintain decay heat removal after failure of the main heat sink to protect not only the concrete structure against excess temperatures, but also the reactor components, especially the reactor pressure vessel. This must be achieved after external events, too.

Due to the inherent safety properties of the modular HTR-2 NPP active heat removal is not required to keep fuel temperatures below fuel design temperature, but to limit the temperatures of the reactor pressure vessel and of the concrete structures forming the primary cavity. With respect to plant design active heat removal may be interrupted for 15 hours.

Our safety assessment has confirmed the design criteria applied for the active cooling systems. Thus we judge the operation modes of the active cooling systems to be adequate and to comply with the applicable safety requirements.

The surface cooler will be arranged about 10 cm above the inner wall of the reactor cavity; its distance to the reactor pressure vessel will be about 1.5 m. It consists of eight bolted sections, which will be suspended in a way that no compulsive forces due to heat extension will occur. The sections will consist of vertical tubes, which are connected by welded bars. The tubes are assigned in an alternating pattern either to redundance 1 and 2 of the secured intermediate cooling system or to the nuclear intermediate cooling system. According to its assignment each cooling tube is connected in its lower part to one out of three headers.

The surface cooler of the modular HTR-2 NPP has to remove about 400 kW during normal operation and about 850 kW after failure of the main heat sink. Each of the three planned cooling trains is designed for the aforementioned requirements.

The coolant flow and reflux can be interrupted by valves arranged within the reactor building. Between the components to be cooled and the isolation valves connecting sockets for fire hoses are foreseen.

The secured intermediate cooling system has to fulfil its safety-related task also during maintenance and simultaneous occurence of a single failure. This means for the two-train redundant system that it must be acceptable to interrupt decay heat removal for a sufficiently long time and to finish the maintenance activities before limiting design values are exceeded.

The modular HTR-2 NPP is characterized by the necessary inherent safety-related features required for the aforementioned procedure; according to our safety assessment active heat removal is not required to avoid excess temperatures above fuel design temperature and cooling of the concrete structures as well as of the reactor components can be interrupted for 15 hours without damage. Thus a twofold redundant system will be sufficient, provided that both redundant trains will not fail due to the same cause. We have taken into account the latter requirement by investigating the mutual interaction between both trains.

After failure of the intermediate cooling systems the system pressure can rise from 5 bar to approximately 20 bar within 15 hours due to heat-up of the surface cooler. Under these circumstances a damage of the surface cooler encompassing both redundant trains of the cooling system due to an assumed failure of one of the surface cooler headers cannot be excluded completely.

In our view this combination of events has an extremely low probability to occur, as the surface cooler including its headers will be designed for at least 40 bar and thus distinctly above system pressure in the discussed incident. Furthermore the initiating incident - simultaneous failure of all intermediate cooling trains - is classified as a design basis accident having a very low probability to occur.

2.2.3 Safety Enclosure

The safety concept of the modular HTR-2 NPP does not provide for a pressure-resistant and gastight containment to enclose released radioactive nuclides. Instead, the planned safety enclosure shall facilitate activity control in accidents accompanied by radioactivity discharge to the environment and serves to minimize the radiation exposure. The safety enclosure consists of

- the reactor building,
- the installations for building relief and ventilation system isolation,
- a subatmospheric pressure ventilation and filter system.

This design provides for separate procedures to meet the dominant protective requirement. For this reason the applicants have designed the safety enclosure referring to two different assumed courses of event, which are described in the following.

During a primary circuit leakage with a cross-sectional area corresponding to a measurement pipe having a maximum diameter of 10 mm, no excess pressure within the reactor building as compared to the environment will build up. The accident will be indicated by the room activity monitoring system. The applicants are planning to ventilate the

building by the subatmospheric pressure ventilation system equipped with aerosol and activated charcoal filters.

In the case of a postulated rupture of a pipe (∅ 65 mm), which is joined to the primary circuit and cannot be closed, an excess pressure will build up in the reactor building. To reduce pressure the primary coolant will be lead to the environment via relief channels and the ventilation stack. Under normal operation conditions the relief channels are isolated from the ventilation stack by ventilation valves, which open at a pressure of 1.1 bar and close automatically after pressure balance will have been established. Additionally each relief channel is equipped with an isolation valve, which is open under normal operation conditions and will be closed by manual initiation after a pressure relief accident, if the automatic relief valve does not close. Thus a controlled air flow within the reactor building can be maintained by means of the subatmospheric pressure ventilation system. The applicants are planning to operate this ventilation system and its filter equipment also during heat-up of the reactor core after the rupture of a 65-mm-pipe to minimize radioactivity discharge to the environment.

For justification of the safety enclosure concept the applicants state that due to the favourable activity retention properties of the fuel, core and reactor design as well as utilization of helium as coolant no specific requirements concerning the safety enclosure appear to be necessary. They emphasize that the results of their radiation exposure calculations demonstrate that even an unfiltered ventilation of the reactor building during core heat-up after rupture of a 65-mm-pipe will not cause excess doses above the dose limits according to § 28 sec. 3 of the Radiological Protection Ordinance.

We agree to the applicants that the design of the safety enclosure should take into account the favourable safety properties of small high temperature reactors. The most important safety feature in assessing the safety enclosure of the modular HTR-2 NPP is the low activity release from the fuel under normal operation conditions and during accidents. With reference to the results of our safety assessment we confirm that the radiation exposure in the environment after accidents with radioactivity discharge to the environment will be below the dose limits according to § 28 sec. 3 of the Radiological Protection Ordinance, even if the effect of the planned filters is not taken into account. Thus it can be justified to abandon a pressure-resistant containment; therefore, filtering of the activity released during core heat-up will only be important with respect to the minimization principle of the Radiological Protection Ordinance.

The reactor building is designed for the pressure loads accompanying the postulated events. The rate of flow of the subatmospheric pressure ventilation system is designed to maintain the required depression as compared to the environment after pressure balance has been established, taking into account the specified reactor building leakage. As a prerequisite to minimize the radiation exposure in the environment by controlled activity

discharge pressure relief must occur reliably in accidents with excess pressures in the reactor building. Otherwise the activity would be discharged near the ground passing through building leakages in such an event. According to present design the pressure relief valves will open in all accidents causing a pressure of at least 1.1 bar in the reactor building. However, in our view pressure discharge acciddents due to medium-size leakages appear to be possible, where pressure in the building will be below the initiation limit of the relief valves. For this reason we see the necessity to analyse this class of incidents in system detail planning and to demonstrate which measures will be taken to prevent radioactivity discharge near the ground.

2.2.4 Electrical Installations

Under normal operation conditions the electrical installations of the modular HTR-2 NPP have to supply the electrical equipment of the systems as well as the instrumentation and control devices with power and to lead the electrical power generated by the plant to the high voltage grid.

In the course of accidents the electrical installations have to supply the safety installations with power required for accident management. For this reason the electrical equipment of the safety installations is connected to the emergency power supply system, which consists of two separated trains as the cooling systems. Apart from this the applicants are planning a further one-train emergency power supply system in the emergency control room of the reactor building.

The components connected to the emergency power supply system will be supplied with power predominantly by the auxiliary power system of the plant. After failure of the auxiliary power system two emergency diesel engines, each of them assigned to one train, are available for power supply. Each of the diesel engines is designed to supply the electrical components required for accident managment with power.

The design of the emergency power supply of the modular HTR-2 NPP complies with the requirement that after failure of the auxiliary power supply and simultaneous non-availability of both emergency diesel engines the plant must be kept within its design limits for up to 15 hours. The safety-relevant instrumentation and control equipment and the control room will de supplied with power by the batteries of the 220-V-DC system for up to 2 hours. The measuring devices and other electrical components connected to the emergency control room will be supplied by the 24-V battery installed in the emergency control room for up to 15 hours.

The applicants classify an even longer grid failure and simultaneous non-availability of both diesel engines as a hypothetic event due to its low probability to occur. However,

they will equip the plant with an external connection for power supply of the emergency control room to reduce the remaining risk in this event beyond design basis.

Accident management for at least 15 hours to keep the plant within its design limits in the case of a complete failure of all active safety installations - even neglecting possible power supply by the emergency diesel engines - will be sufficient to cover the longest breakdown durations of the public grid.

In our view this safety property of the modular HTR-2 NPP is the relevant feature to justify an emergency power supply consisting of only two diesel engines. From a deterministic point of view a two-train emergency power supply would be considered to fail taking into account the single failure criterion and a potential repair fault. With respect to the well-known breakdown durations typical for the public grid the aforementioned combination of events is not safety-relevant for the modular HTR-2 NPP due to its response to power supply failures.

In our opinion, however, an estimated very low probability of incidents with grid failure times of more than 15 hours does not justify to consider them as exclusively hypothetical events. From a probabilistic point of view we conclude that the emergency power supply must be available with sufficient reliability not later than 15 hours after begin of the event to ensure compatibility with overall plant design. This means for the diesel engines and other required equipment that their reliability required for accident management is not determined by the need for an immediate availability, but that they can be repaired or taken into operation within a relatively large time interval. As a prerequisite to proceed as described, we see, however, the need to observe the relevant quality assurance requirements of the KTA-rules in planning and inspecting the emergency diesel engines and the emergency power distribution gear.

If the aforementioned aspects are considered sufficiently and correctly, we will agree that the long-term failure of the emergency power supply due to the simultaneous occurence of the incidents "grid failure" and "long-term unavailability of the emergency power supply" can be classified as a hypothetical event.

2.2.5 Reactor Protection System and Emergency Control Room

The control room is located in the switch gear and emergency supply building and contains the operation, information and signaling equipment required for operation and surveillance of the plant. After failure of the control room surveillance of the plant will be monitored in the emergency control room located in the reactor building, which is designed for all external incidents. In the emergency control room all data are signaled and recorded, which characterize the safety state of the plant and provide the information

required for the necessary steps in accident management. Further radiological and meteorological data are signaled to indicate a possible radioactivity discharge to the environment and to determine the activity propagation conditions.

The emergency control room can be entered through a separate external entrance, if access from the switch gear and emergency supply building is impossible.

Apart from the small-sphere shut-down system no further system can be initiated from the emergency control room. If the control room is not available, the need for possible manual actions can be recognized evaluating the data signaled in the emergency control room. The necessary actions can be initiated locally by means of control devices.

The emergency control room including the measurement equipment is supplied with electrical power by a one-train emergency power grid. To ensure the required power supply after a possible failure of the emergency power supply the emergency control room is equipped with a battery designed for operation up to 15 hours. After this time interval power supply can be maintained by a mobile emergency power generator using a planned cable connection, until grid power supply is reestablished. It is assumed that the mobile emergency power generator can be supplied by an external organization as the fire brigade.

The initiation of the small-sphere shut-down system being the only active measure to be carried out from the emergency control room results from the safety concept of the modular HTR-2 NPP: Due to the inherent safety features of the reactor automatic actions are not required for accident management, after the reactor protection measures have been initiated at the beginning of the incident. Even after an external event destroying the switch gear and emergency supply building partly or completely the initiation of the small-sphere shut-down system is not required for immediate accident management, but for transition of the reactor from the state "hot, below criticality" to the long-term safe state "cold, below criticality".

Although according to the present status of design further actions initiated in the emergency control room are not required, we will not exclude the need for such actions emerging in the course of detailed planning. These possible necesseties would not affect the overall concept.

The reactor protection system is part of the instrumentation equipment and belongs to the safety system of a reactor; it is designed to initiate automatic actions in the course of accidents. These actions shall ensure that the dominant protective requirements defined in the safety criteria for nuclear power plants are met and that the plant will be kept within its design limits, until manual actions will facilitate long-term accident management. For the modular HTR-2 NPP this overall task reduces to initiate the planned protective ac-

tions only once after the accident has been detected. Active measures aiming at system control during the course of accident, e.g. to ensure decay heat removal or to perform level control in coolant vessels, are not required.

Within its task the reactor protection system has to determine specific safety data, to link safety parameters deduced from these data and to develop initiation signals, which have priority over any other control signal. The accident-specific data, the initiation criteria based on them and the initiation signals developed by the reactor protection system are based on the results of accident analyses.

The reactor protection system determines the safety data

- neutron flux,
- hot gas temperature,
- cold gas temperature,
- moisture content in the primary circuit,
- pressure in the primary circuit,
- pressure in the secondary circuit,
- primary coolant flow,
- feed water flow,
- steam flow.

They are applied either directly or after linking them in calculation circuits to gain safety parameters, from which initiation criteria are developed after well-defined limits have been exceeded.

All initiation criteria will trigger the following protective actions:

- Insertion of the reflector rods,
- shut-off of the primary circuit blower,
- isolation of the steam generator.

Further protective actions depending on kind and course of accident are

- primary circuit isolation,
- steam generator relief.

The steam generator relief valves are closed without being triggered by the reactor protection system after pressure balance has established between primary and secondary circuit.

Normally the accidents are detected by determining at least two physically diversified safety data. If only one of them is available, the KTA-rules require the initiation level to be designed more sophisticatedly. The applicants are planning such a design for detection of steam generator heating tube leakages and steam duct ruptures.

The planned redundant arrangement, the local separation and the constructional details of the equipment to initiate protective actions will ensure sufficiently that even in the case of failure-initiating incidents within the plant or system as well as external events the necessary protective actions to control an accident will be initiated reliably.

The constructional details of the reactor protection equipment and the local separation of redundant components to turn off the primary circuit blower and to insert the reflector rods are of dominant importance with respect to possible partial damage of the switch gear and emergency supply building due to aircraft impact or external shock wave. In our view the planned design - characterized by multiple initiation and initiation-directed failure behaviour of the logic module of the reactor protection system as well as of the switching equipment - is suitable to ensure the required initiation safety even in the case of partial damage of the switch gear and emergency supply building.

The three-train reactor protection system is connected to the two-train power supply system. This means that already the simultaneous occurence of a single failure-initiating incident and an accidental failure can interrupt the power supply of the complete reactor protection system. Due to the initiation-directed failure behaviour of the reactor protection system all reactor protection actions will be initiated, i.e. the reactor will be shut down safely.

2.2.6 Buildings

In our safety assessment of the buildings for the modular HTR-2 NPP we have verified, if the safety requirements are met, which are defined e.g. in the safety criteria for nuclear power plants and the RSK-guidelines and exceed those valid for conventional buildings. Specific requirements apply to buildings, which are necessary either - e.g. due to their constructional design, shielding or barriers - directly or - e.g. due to installation and load design of safety-relevant systems - indirectly

- to shut down the reactor safely and to keep it in the shut-down state,
- to remove the decay heat and
- to ensure safe enclosure and shielding of the radioactive inventory.

The following buildings are classified as safety-relevant:

- Reactor building (UJA),
- reactor building annex (UJH),
- reactor auxiliary building (UKA),
- switch gear and emergency supply building,
- secured cooling cells (URB),
- cable tunnels from UBR to UJA.

We have verified, if these buildings are designed sufficiently to bear the loads resulting from normal operation of the modular HTR-2 NPP and from accidents.

Based on our safety assessment we have recommended inter alia that the ventilation stack to be erected on the reactor auxiliary building should be designed for seismic loads, provided that subsequent damaging of the latter building endangering its stability required for accident management cannot be excluded. We have already discussed our results concerning the reactor building as part of the safety enclosure elsewhere.

Further we have verified, if the arrangement of the buildings described in the safety analysis report is such that damage can be excluded due to mutual impact, e.g. caused by assumed failure of high-energy components, as turbine or vessels with high energy content, or due to fragments resulting from collapsing buildings during an earthquake. The planned arrangement of the buildings will meet the aforementioned requirement. However, the effect of possible mutual impact will have to be verified again, if the exemplary arrangement of the buildings described in the safety analysis report is modified.

3 Accident Analysis

3.1 Spectrum of Incidents

According to § 7 sec. 2 of the German Nuclear Energy Act the necessary precautions against damage from construction and operation of nuclear facilities have to be taken with respect to the state of science and technology. Especially the activity discharge to the environment must not cause radiation doses exceeding the dose limits given in § 28 sec. 3 of the Radiological Protection Ordinance.

To demonstrate that this licensing prerequisite is met the applicants have elaborated a compilation of the HTR modular reactor specific design basis accidents by applying analogically the basic principles of the so-called "Accident Guidelines" /5/ developed for pressurized water reactors. According to the procedures defined in the accident guidelines they have analysed those accidents relevant for the plant, developed design requirements based on the results of the analyses concerning buildings, components and systems and defined radiologically relevant accidents to be analysed with reference to the dose limits given in § 28 sec. 3 of the Radiological Protection Ordinance.

They further have evaluated accidents beyond design basis to illustrate the inherent safety margins of the plant for this class of incidents and to demonstrate that risk-reducing measures have been taken to the required extent.

Due to our safety assessment the applicants have modified and completed their original accident compilation, taking into account these modifications in their revised safety analysis report. In our view the present compilation is complete and defines the design basis of the modular HTR-2 NPP with respect to the German Nuclear Energy Act and the presently common licensing procedures accurately and to the necessary extent.

3.2 Analysis of Accident Progress

Apart from verifying the completeness of the accident-initiating incidents analysed by the applicants we assessed accident progress for the different incidents and evaluated the consequences resulting from the analyses.

In our assessment we applied the deterministic requirements defined in the applicable rules and guidelines or adapted to the features of the modular HTR-2 NPPP, respectively. Therefore the originally submitted accident analysis had to be revised.

In their review the applicants have based their accident analysis on the usual unfavourable assumptions typical for nuclear licensing. These assumptions are e.g.

- postulated failure of the first initiating criterion for the reactor protection system,
- unfavourable initial conditions in the system under consideration,
- single failure and - when applicable - non-availability due to maintenance of the systems required for accident management,
- neglecting operational systems for immediate accident management.

This modified procedure caused the applicants to provide for additional safety installations, e.g. a limitation of rod insertion, and additional initiation criteria for the reactor protection system, e.g. the maximum steam temperature, as well as to modify safety-relevant limits, e.g. fuel design temperature. Consequently, the applicants had to revise existing analyses or to submit new ones.

In the following sections we will summmarize some important results of the accident analyses.

3.2.1 Reactivity Accidents

The applicants have investigated reactivity accidents initiated during normal operation of the equilibrium core and performed parameter variations to analyse the influence of unfavourable operation modes. They have taken into account accident-aggravating single failures, as e.g. erroneous start-up of the primary circuit blower and maloperation of the small-sphere shut-down system.

Our investigations concerning first reactor core, zero power operation and the start-up incident as well as malfunction of single reflector rods confirm that the accident "withdrawal of all reflector rods at maximum rate during full power operation of the equilibrium core" is to be considered as a covering design basis accident.

In this accident reactor scram required to limit the maximum temperatures of the pressure vessel unit is initiated by two physically different criteria. The reactor will be shut down safely despite the postulated failure of the first initiating criterion, even under the aggravating assumption that one reflector rod will not drop. Temperature design limits will not be exceeded; unacceptable power excursions will not occur.

Apart from the effects of absorber withdrawal we have analysed the reactivity-influencing incidents

- erroneous start-up of the primary circuit blower,
- incident-induced reduction of the cold-gas temperature,
- water leakages to the primary circuit,
- densification of the pebble bed due to earthquakes.

Also in these cases our results confirm that the possible effects are covered by the design basis accident investigated by the applicants. However, in our safety assessment report we have made some hints and given several recommendations concerning different limiting conditions to be established, as e.g. maximum power limitation to 105 % of rated power and avoidance of an unplanned start-up ot the primary circuit blower at high coolant gas temperatures, which will result in the reactor state "hot, below criticality" and simultaneous removal of the decay heat via the surface cooler. We further have pointed out in our report that additional investigations are required concerning the earthquake-induced densification of the pebble bed, where transfer of experiments carried out for the THTR reactor to the modular HTR-2 NPP is of special interest. The HTR-module core differs from the experimental set-up of the THTR model with respect to core geometry and construction of the core components; this can amplify the induced oscillations and prolong their duration, the resulting effect has to be investigated both analytically and experimentally.

3.2.2 Disturbed Heat Removal Without Coolant Loss

Disturbed heat removal without coolant loss will be caused by the incidents

- interruption of primary coolant flow,
- disturbed steam removal,
- disturbed feed water supply,
- failure of auxiliary power supply.

These incidents will cause deviations of thermodynamic parameters from their normal operation values. The deviations will be indicated by the limitation eqipment and reactor protection system, which will initiate the necessary protective actions.

We want to point out that the applicants have based the HTR modul design on the assumption that electrical power supply will fail totally in these incidents, i.e. external grid supply and auxiliary power supply will fail and emergency power supply will not be available. These assumptions deviate markedly from presently common procedures in nuclear licensing.

The applicants distinguish the following scenarios:

- Short-term failure of the auxiliary power supply system
  (less than or up to 2 hours),
- medium-term failure of the auxiliary power system
  (less than or up to 15 hours),

- long-term failure of the auxiliary power supply system (more than 15 hours).

In the case of short-term failure the plant is shut down as in normal operation by means of the main heat sink after power supply from the grid has been reestablished. Design limits will not be violated.

During a medium-term failure the plant is in a safe state, i.e. the reflector rods are inserted, the primary circuit blower is turned off and the steam generator is isolated. Two hours after the initiating incident the control room is not available as its power supply is interrupted. Due to the initiation-directed failure behaviour of the reactor protection system the primary circuit is locked and the steam generator is relieved. During core heat-up the primary circuit safety valve can open. After auxiliary power supply has been reestablished within 15 hours after the initiating incident the surface cooler can be operated again and the plant is shut down.

This incident was analysed extensively. The results indicate that locally the temperature of the primary cavity concrete in the reactor building is at its design limit of 150 °C after 15 hours. Within this time interval the maximum temperature of the reactor pressure vessel amounts to 310 °C; that of the surface cooler reaches 220 °C, i.e. it can be operated after reestablishing power supply. The maximum fuel temperature does not depend on operation of the surface cooler, it will be below 1200 °C.

Our results indicate that even a complete failure of the surface cooler with the reactor kept in its pressurized state for 15 hours will not cause unacceptable loads of the reactor pressure vessel including its components, of the reactor building and of the surface cooler. The radiation exposure in the environment due to discharge of primary coolant will be far below the dose limits according to § 28 sec. 3 of the Radiological Protection Ordinance.

The applicants do not take into account the complete failure of power supply for more than 15 hours in their design basis analyses. In our view this can only be accepted with respect to the two-train design of the emergency power supply, if a short-term repair of at least one emergency power generator can be performed reliably. We have defined the respective requirements in our safety assessment report, including a demand for adequate quality assurance measures.

3.2.3 Loss-of-Coolant Accidents

In the following we distinguish between ruptures and leakages in the primary and secondary circuits as well as ruptures of steam generator heating tubes with special emphasis on their effects concerning

- maximum fuel and component temperatures,
- loads, e.g. differential pressure loads on the reactor building,
- radiation exposure.

As ruptures are excluded for the pressure vessel unit itself primary ruptures are to be assumed only for pipes connected to the pressure vessel unit. These pipes have either a maximum diameter of 65 mm or the potential leakage areas are reduced to a corresponding cross-section by constructional measures as in the case of the fuel element discharge pipe.

The rupture exclusion for the pressure vessel unit can be ensured largely by observing the principles defined in the frame specification "Base Safety" /6/, which have been applied successfully in light water reactor technology. In our view additional measures are required with respect to the high-temperature materials and their weldings close to the steam duct socket and to material embrittlement due to neutron irradiation at comparatively low operation temperatures. In our safety assessment report we have accounted for the need of an operation-preceeding investigation program and of operation-accompanying irradiation samples. We confirm on the basis of our fracture-mechanical analyses that ruptures of the pressure vessel unit can be excluded. However, a constructional modification of the steam pipe socket turned out to be necessary, as a planned welding between two different materials was identified as critical and therefore was shifted from the high-temperature area. Thus we confirm the applicants' rupture assumptions for the pressure vessel unit (see sec. 2.2.1).

Primary ruptures and leakages must be analysed in different groups:

- Ruptures of large connecting pipes,
- ruptures of small pipes or small leakages, respectively,
- ruptures or leakages of primary coolant containing pipes outside the reactor building.

The double-sided rupture of the largest connecting pipe with a diameter of 65 mm containing primary coolant, which cannot be closed due to its position close to the reactor pressure vessel, proved to cover the effects of all other possible primary ruptures.

We have verified by independently performed calculations taking into account unfavourable initial conditions and both systematic and statistical uncertainties of the relevant pa-

rameters that the acceptable maximum fuel temperature will not be exceeded, i.e. the fuel temperature will be below design limit.

Further our calculations indicated that the isolation in the lower part of the reactor pressure vessel must be modified to avoid excess temperatures above design limit taking into account uncertainties of the temperature-influencing parameters. The design temperature of the reflector rods amounting to 650 °C will be exceeded markedly, although this will not affect their function and integrity. We consider this result not to be relevant for the overall design.

The radiation exposure in the environment caused by the pressure relief accident - even taking into account the subsequent core heat-up and a postulated failure of the filtering system - is far below the dose limits according to § 28 sec. 3 of the Radiological Protection Ordinance.

In our safety assessment report we have defined requirements concerning the automatic filtering of the activity discharges due to small leakages and we have justified in detail the need for further investigations concerning medium-size leakages, which will increase the pressure inside the building to values below the initiation level of the pressure relief valves. Again, these results are not relevant for the overall design.

As a result of our investigations concerning secondary ruptures and leakages, e.g. ruptures of the feed water or steam ducts, the concept to interrupt steam removal had to be modified to keep the pressure inside the reactor building for a two-module plant within the planned design limits. We have calculated independently from the applicants pressure increases for each of two rupture positions covering all possible primary and secondary ruptures. Our results confirm the design excess pressure of the reactor building amounting to 0.3 bar. However, this value is valid for the reactor hall and the largest part of the reactor building outer walls only. Parts of the building close to the rupture location will be charged with distinctly higher differential pressures, which have to be taken into account as special loads and combined with other loads in static design.

Our analyses of water leakages into the primary circuit due to steam generator heating tube leakages indicated that design and operational measures will ensure that the leakage size will be limited to the cross section of a single heating pipe. Thus we confirm the water inventory of 600 kg to be conservative, which the applicants have determined to penetrate into the primary circuit. The initiation of protective actions by only a single initiation criterion developed from the moisture measurement can be accepted, if that part of the reactor protection system relevant for this classs of incidents will be designed sophisticatedly according to the KTA-rules. Further we have investigated the influence of steam ingressed into the primary circuit on reactivity behaviour of the core and corrosion

of the fuel elements accompanied by formation of water gas. Based on our results we confirm that safety-relevant limiting values are not exceeded.

3.2.4 External Events

We have investigated the influence of external events on safety of the modular HTR-2 NPP according to the usual methodology in nuclear licensing. Based on our results we have demanded that the extent of plant components and buildings designed for earthquake had to be enlarged. The applicants are planning to generate the basic seismic load assumptions by means of a new, empiric-statistical method, which aims at deducing the seismic-structural design data from a probabilistic analysis of seismic hazard. We have verified that the planned procedure is a consistent method providing realistic seismic load data. However, we have recommended that for a concrete site of the modular HTR-2 NPP the seismic structural design data should be generated additionally by means of the proven deterministic method.

Our analysis of further natural external events did not result in any requirements to modify plant design.

3.3 Risk-Reducing Measures

The events "aircraft impact" and "external shock wave" are characterized by an extremely low probability to occur. For this reason their possible effects are not classified as design basis accidents in the accident guideline for pressurized water reactors. Thus the design characteristics to be met for these events aim at reducing the risk due to operation of the plant.

According to present plant design the reactor building and safety-relevant components and systems within this building shall be designed for loads from aircraft impact and an external shock wave. In compliance with the RSK guideline /6/ the loads due to an aircraft impact are assumed to be independent from the site.

The design of the reactor building and the safety-relevant components and systems for an external shock wave is based on the standard pressure-time graph given in the BMI guideline for design of nuclear power plants against shock waves from chemical reactions /7/. If due to site-specific features, e.g. industrial plants in close neighbourhood, shock waves inducing higher loads appear to be possible, design will be based on a site-specific pressure-time graph of the shock wave.

An aircraft impact as well as a shock wave will destruct partially or completely the switch gear and emergency supply building. This can affect the function of the reactor protection system and emergency power supply system to such an extent that both systems will fail.

The applicants are planning to design the reactor protection system such that the protective actions

- insertion of the reflector rods,
- turn-off of the primary circuit blower,
- isolation of the secondary circuit,
- isolation of the primary circuit,
- steam generator relief

will be initiated when necessary due to plant behaviour or as a result of damage to the reactor protection system itself.

Surveillance of the plant and long-term control of subcriticality will be performed in the emergency control room, which is located in the reactor building designed for the aforementioned events. The emergency control room is unrestrictedly accessible and has a separate external entrance.

The emergency control room is equipped with a battery-supported one-train emergency supply system to ensure its function and to supply peripheric systems with power (ventilation, lighting and communication systems). The battery is designed for operation up to 15 hours. After this time interval power supply will be maintained by a mobile energy supply engine using a special cable connection until grid power supply will be reestablished; the energy supply engine will be provided by the fire brigade or another institution. When necessary, two sections of the surface cooler can be operated by connecting them to fire hoses joined externally to the intermediate cooling system.

The possible interruption of decay heat removal for at least 15 hours in conjunction with the independent emergency power supply of the emergency control room grant a sufficient time interval to perform measures to reestablish heat removal by means of the surface cooler.

Thus we confirm on the basis of our design review that the extent of risk-reducing measures for the modular HTR-2 NPP by protecting it against civilization-induced external events and by providing steps to establish an external feedwater supply of the surface coolers as well as power supply of the emergency control room meets the applicable requirements and takes into account the plant-specific properties sufficiently.

4 Radioactive Substances and Radiation Protection Measures

4.1 Radiation Protection and Radiation Exposure of the Personnel

The assessment of plant design with respect to radiation protection of the personnel shall ensure that planning measures have been taken

- to avoid any unnecessary radiation exposure and contamination and
- to keep any unavoidable radiation exposure or contamination not only below the limiting values given in the Radiological Protection Ordinance, but as low as possible

according to the requirements of the Radiological Protection Ordinance.

There will be different radiation sources in the modular HTR-2 NPP: The prompt nuclear radiation and the predomimnant part of fission product radiation stem from the nuclear processes in the reactor core. The γ-radiation due to electron capture and the activated nuclides will also be generated by neutron radiation outside the reactor core. Finally, activated nuclides and a small part of the fission products can be transported by the coolant to systems and components adjoined to the primary circle.

We have verified, if the applicants have taken into account the relevant radiation sources in plant design and if the source terms can be applied to generate realistic radiation fields and levels to be expected in the plant. Based on our results we confirm that the applicants' methodology is suitable to generate the required data and the information given in the safety analysis report and further design documents is consistent and complete.

The radiation field of the reactor core will be shielded in axial and radial direction by different components, which as a whole form a sandwiched structure. The inner layer will mainly consist of the graphite of the fuel elements, the borated carbon material parts and the reactor pressure vessel. The outer layers will be made from concrete and form the biological shield. Apart from the reactor core especially the following components and systems have to be shielded due to their activity inventory:

- Fuel handling equipment,
- helium purification plant,
- liquid waste treatment equipment,
- storage facility for radioactive waste.

The basic material of all shielding walls is concrete. Further shielding materials are steel and lead, e.g. for shielded doors or local shieldings, as well as special concrete at all places, where due to lack of space standard concrete would be insufficient as shielding material.

The modular HTR-2 NPP does not diiffer from other nuclear power plants with respect to the basic design principles of shielding. The planned shielding concept and the radiological protection measures described in the following are suitable to protect the personnel against the hazards of radioactive radiation according to the state of science and technology. If local measurements indicate the need for additional shielding measures, the space to install further shielding equipment appears to be sufficient. The applicants have to plan the radiation shielding measures in detail within constructional design of buildings and components.

Apart from the shielding concept further measures are planned to protect the personnel of the modular HTR-2 NPP against the hazards of radioactive radiation:

- Separation of nuclear and conventional components and systems,
- separation of high- and low-active components in nuclear systems,
- separation of individual high-active components with respect to maintenance and repair,
- local separation of components, valves and operating stations on the one hand and of internal passages for the personnel on the other hand.

The planned arrangement of rooms and components will contribute extensively to reduce the radiation exposure of the personnel. Examples are:

- Both modular reactors will be erected separately in primary cavities,
- the pumps of the secured intermediate cooling system will be arranged in different rooms of the reactor building annex and will be shielded,
- the containers for concentrated liquid waste will be erected in separate, shielded rooms of the reactor auxiliary building.

The concept to mutually shield activity-containing components will be realized to a large extent in the controlled area. This does not apply to both waste water vessels, which will be arranged in a common room. We recommend to arrange these components in separate rooms.

According to the classification system of radiological protection areas described in the safety analysis report the reactor building and the predominant part of the reactor auxiliary building are part of the controlled area.

The rooms containing the steam generator and the reactor pressure vessel are inaccessible.

The applicants are intending to classify the rooms within the controlled area with respect to local dose rates. The upper limits of the individual classes roughly differ by a factor of

ten. This classification system shall facilitate accessibility even to areas with high local dose rates by keeping the dose rates along the access passages at low values. Thus all rooms within the reactor building apart from the primary cavities will be accessible during operation.

As a result of the radiological protection measures planned by the applicants the dose rates outside the controlled area will be below dose limits applicable to operational surveillance areas and outside the power plant area below those for non-operational surveillance areas.

We agree to the planned concept to install distinguished radiation protection areas and to classify the rooms within the controlled area according to local dose rates; during future datail planning the different rooms can be assigned to radiological room classes. When necessary, further constructional measures to improve shielding will be possible. Our independently carried out calculations confirm the assignment of the operational and non-operational surveillance areas performed by the applicants. In our view there will be no need to install a non-operational surveillance area, if the plant area is separated from the neighbouring areas in an appropriate manner, i.e. the distance to the plant fence is sufficiently large.

In additionally submitted documents the applicants have described in detail several important maintenance activities and regular periodic inspections to demonstrate how the requirements of the guideline on precautionary measures in radiation protection /8/ will be met: The applicants are planning to apply special tools or measures to reduce the required time for maintenance or repair of primary circuit components or others in adjoining systems, e.g.

- camera assistance in preparing reactor pressure vessel inspections and preceding automized ultrasonic testing,
- application of easy-to-detach isolations,
- stud tensioning device for reactor pressure vessel closure head,
- mobile shielded working platform above the upper thermal shield.

The basic requirements concerning precautionary protective measures for regular periodic inspections as well as for maintenance and repair especially of the reactor pressure vessel are met by the intended measures. However, the detailed planning of the working procedures accompanying future design and construction steps must be based on a break-down of the collective doses with respect to separate contributions to individual doses and a detailed description of the working sequence taking into account local dose rates.

4.2 Discharge of Radioactive Substances During Normal Operation and Radiation Exposure in the Environment

The applicants have described the processes, the applied calculation models and the extent of activity release from the fuel elements to the primary coolant in the safety analysis report. Several phenomena contribute to activity release to the coolant:

- Activity release from intact particles,
- activity release from particles damaged during manufacture,
- activity release due to radiation-induced damage,
- activity release due to contamination of the graphite matrix.

The intact coatings of the particles will form an efficient barrier against activity release, i.e. intact particles will not contribute relevantly to release rates.

However, during manufacture sporadic damaging of single particles cannot be excluded completely, i.e. the relevant barriers for activity retention will not be effective. In these cases the activation and fission products will migrate from their origin in the fuel grain to the grain boundaries by diffusion, followed by grain boundary diffusion to the graphite matrix and finally to the fuel element surface.

Release of gaseous fission products will only be retarded by the slow diffusion process in the fuel grain. Grain boundary diffusion and diffusion in the graphite matrix are relatively fast processes. Diffusion of non-volatile fission and activation products both in the fuel kernels and in the graphite matrix is a relatively slow process with the effect to retard release of these nuclides and thus to reduce the release rates especially of the short-lived isotopes. Design calculations are based on an assumed part of defect particles (expected value), which doubles that verified in specific investigations.

In principle, additional particle defects, induced e.g by burn-up, fast neutron fluence or temperature loads, can occur during operation of the fuel elements in the reactor core. The applicants have deduced the expected part of irradiation-induced particle defects from irradiation experiments. Design is based on a defect rate assumed conservatively to be higher than the experimental results by a factor of ten.

The natural graphite contained in matrix graphite contains inter alia traces of uranium stemming from natural contamination. Thus, apart from activation of further contaminants, fission of uranium will occur to a minor extent in the graphite matrix outside the kernels. The resulting fission and activation products can migrate through the graphite matrix to the fuel element surface by means of the aforementioned transportation mechanism. The applicants are combining the inventories of uranium due to matrix contamination and to fabrication-induced particle defects to a so-called "free uranium inven-

tory" to specify the acceptable maximum uranium content in the matrix and to calculate the release rates. The design value for the fabrication-induced particle defects includes the uranium contamination of the graphite matrix.

In the following we summarize the results of our evaluation concerning the release rates calculated by the applicants:

- The break-down of the source terms into fabrication-induced and irradiation-induced particle defects as well as into matrix contamination and the different release models deduced from these mechanisms appear to be reasonable,
- the physical models applied in the calculations describe the relevant transportation phenomena of fission and activation products and take into account the state of science and technology,
- the input data, e.g. particle defect rates, diffusion coefficients and other material data, appear to be sufficiently conservative for the calculations,
- the nuclide vector has been determined with respect to the radiologically relevant isotopes.

Our calculations of the activity inventory based on conservatively estimated release rates indicate for some nuclides distinctly lower release rates as compared to the applicants' results, whereas we confirm the release rates calculated for the remaining nuclides. Thus we expect that the release rates calculated by the applicants will cover those to be expected during normal operation.

The radioactive nuclides released to the coolant or generated there by activation, respectively, are transported by the coolant from the core to the primary circuit. In normal operation a quasi-steady-state coolant activity will result, which can be calculated performing a balance of all source and loss terms.

We have already dealt with the most relevant source term due to activity release from the fuel elements. To a minor extent aerosole nuclides stemming from radioactive decay of the short-lived noble gases contribute to the coolant activity, too.

The relevant loss terms are governed by radioactive decay, the filtering effect of the helium purification system and losses via primary circuit leakages. Additionally plate-out of radionuclides on the inner surface of the primary circuit will reduce the aerosole activity in the coolant.

In our safety assessment we have verified the design data of the primary coolant activity given by the applicants and performed some independent calculations. The nuclide-specific results are listed in our safety assessment report and confirm the design data.

Further we have determined and evaluated the activity inventories in the auxiliary systems, e.g. in the helium purification system, starting from the primary coolant activity. The results have been presented in our safety assessment report. In the following we are giving some comments concerning the possibility to operate the secondary circuit in an "unclosed" mode, i.e. to use part of the steam for process purposes.

The applicants have applied for a tritium concentration in the process steam, which meets the requirements of § 4 sec. 2 of the Radiological Protection Ordinance. In conjunction with attachment III sec. 2 of the Radiological Protection Ordinance radioactive substances can be applied provided that their specific activity is less than 100 Bq/g.

As the applicants are planning to use process steam outside the power plant and thus outside the surveillance area, the condensate of the process steam must be reusable without any restriction. If in future licensing steps no annual limit for the yearly discharge of tritium with process steam is established, the requirements of § 46 sec. 4 will have to be met, according to which process steam condensate can only be drained off to public sewers or rivers, lakes, and canals, if the waste water activity is not greater than 1.25 times the value given in attachment IV, tab. IV 1, column 6 of the Radiological Protection Ordinance.

If the use of condensate as drinking water cannot be excluded, the acceptable maximum tritium uptake per year according to § 46 will be exceeded slightly on the basis of the planned tritium concentration. This aspect has to be clarified before the corresponding licence is issued.

The applicants are planning to verify by sampling that the discharges to the process steam are kept below the limits given in the licence. If sampling is sufficient or a continuous surveillance is required, depends on the planned use of the process steam and has to be decided during future licensing steps.

For verification of the room activity concentrations in the reactor cavities we have investigated the mechanisms

- activation of the air close to the reactor pressure vessel,
- activation of the metallic surfaces of the reactor pressure vessel and the surface coolers,
- primary coolant leakages.

We confirm the applicants' data on these mechanisms to be conservative.

Further we have verified the activity discharges during normal operation of the modular HTR-2 NPP, based on the acitivity concentrations in systems and rooms, and listed the

results in our safety assessment report. Our nuclide-specific results refer to discharges with

- exhaust air from the ventilation stack and the turbine building,
- waste water from the liquid waste treatment system and from secondary circuit leakages as well as
- solid radioactive waste, as e.g. spent fuel or normal operational waste.

These discharges of radioactive substances will cause a radiation exposure in the environment, which must not exceed the dose limits according to § 45 of the Radiological Protection Ordinance. In licensing this radiation exposure has to be calculated with respect to the most unfavourable receiving points taking into account all relevant exposure pathways and including the food chains. The administrative guideline on the applicaton of § 45 of the Radiological Protection Ordinance /9/ contains calculational models and parameters to determine the radiation exposure.

According to this guideline the human radiation exposure is defined as the radiation exposure of a member of the critical population group and is caused by external irradiation, i.e. the contribution due to externally effective radiation sources, and internal irradiation, i.e. the contribution due to incorporated radionuclides. A member of the critical population group is a person, who is exposed to the maximum radiation exposure due to one or more exposure pathways at the most unfavourable receiving point. The radiation exposure shall be determined on the basis of realistic habits to be assumed for part of the population. Extreme habits, e.g. in food consumption, shall not be considered. The most unfavourable receiving point is that location in the environment of the modular HTR-2 NPP, which due to the distribution of radionuclides discharged to the environment will experience the highest radiation exposure. It is assumed that the consumed food is produced at the location characterized by the highest food contamination in the respective area.

In the safety analysis report and in additional documents the applicants have described their calculations and listed the results of the radiation exposure of adults and infants due to radioactive effluents from the modular HTR-2 NPP with the exhaust air and waste water. Their calculations are based on the discharge data submitted in their application and on the procedures defined in the aforementioned guideline, taking into account all exposure pathways and conservative transfer data.

We have determined in an analogous procedure the radiation exposure caused by radioactive effluents from the modular HTR-2 NPP with the exhaust air and waste water and applied the calculational models and data given in the guideline on the applicaton of § 45 of the Radiological Protection Ordinance. Whenever necessary, i.e. in cases lacking site-specific data, e.g. meteorological data, we have used the same data as the applicants,

which are listed in the safety analysis report. Before applying these data, we have verified, if they are realistic and can be taken as representative for a possible site.

Our calculation of the radiation exposure is based on the discharge data determined in our safety assessment instead of the activity discharge data the applicants have applied for. Thus our results of the radiation exposure are below those of the applicants. However, we confirm that the radiation exposure due to radioactive effluents both with the exhaust air and waste water will be distinctly below the dose limits according to § 45 of the Radiological Protection Ordinance.

We have compiled the radiation exposure determined by us in relation to the dose limits according to § 45 of the Radiological Protection Ordinance in tables 4-1 and 4-2. Deviations from the results of our safety assessment report (October 1989) are caused by a recent review of the Radiological Protection Ordinance and the calculational procedures. However, these deviations are of minor importance only.

The results are distinctly below the dose limits according to § 45 of the Radiological Protection Ordinance. However, for a concrete site the additional radiation exposure due to other nuclear facilities or handling of radioactive substances has to be taken into account. As a concrete site for a modular HTR-2 NPP has not been proposed these contributions to radiation exposure are not considered in the aforementioned tables.

After a concrete site has been chosen, it might become necessary to modify the obtained data, as site-specific features can influence the calculated radiation exposure, as e.g. the flow conditions of the run-off water, meteorological, orographic or settlement conditions in the environment of the plant as well as specific food consumption habits of the population. Thus the calculation of the radiation exposure to be expected under normal operation conditions has to be repeated once more after a site has been determined.

As by now the calculated dose rates are distinctly below the dose limits according to § 45 of the Radiological Protection Ordinance, we expect the dose rates of a concrete site to be within these limits.

The radiation exposure due to direct radiation from the plant is negligible as compared to that due to radioactive effluents with the exhaust air or waste water. Thus it is distinctly below the dose limit according to § 44 of the Radiological Protection Ordinance.

Tissue	Annual dose in µSv		
	Adults	Infants	Dose limit § 45 RPO
Bladder	27	47	900
Breast	28	48	900
Upper large intestine	27	47	900
Lower large intestine	27	47	900
Small intestine	27	47	900
Brain	28	48	900
Skin	29	48	1800
Testes	28	47	300
Bone surfaces	28	48	1800
Liver	28	47	900
Lung	28	47	900
Stomach	28	47	900
Spleen	28	47	900
Adrenals	27	47	900
Kidneys	28	47	900
Ovaries	27	47	300
Pancreas	27	47	900
Red bone marrow	28	47	300
Thyroid	28	49	900
Thymus	28	47	900
Uterus	27	47	300
Effective dose equivalent	28	47	300

Table 4-1: Potential radiation exposure due to activity discharge with the exhaust air

Tissue	Annual dose in µSv		
	Adults	Infants	Dose limit § 45 RPO
Bladder	0.9	0.3	900
Breast	0.8	0.3	900
Upper large intestine	1.0	0.3	900
Lower large intestine	1.0	0.3	900
Small intestine	1.0	0.3	900
Brain	0.8	0.3	900
Skin	0.7	0.4	1800
Testes	0.9	0.3	300
Bone surfaces	0.9	0.3	1800
Liver	1.0	0.3	900
Lung	0.9	0.3	900
Stomach	0.9	0.3	900
Spleen	0.9	0.3	900
Adrenals	1.0	0.3	900
Kidneys	0.9	0.3	900
Ovaries	0.9	0.3	300
Pancreas	0.9	0.3	900
Red bone marrow	0.9	0.3	300
Thyroid	1.0	0.5	900
Thymus	0.8	0.3	900
Uterus	0.9	0.3	300
Effective dose equivalent	1.0	0.3	300

Table 4-2: Potential radiation exposure due to activity discharge with waste water

4.3 Radiation Exposure after Accidents

The spectrum of light water reactor accidents to be analyzed radiologically is defined in the accident guideline. According to this guideline the accident analyses have to be based on sufficiently conservative assumptions, calculational models and input parameters to describe the cours of the accident, radioactivity discharge to the environment and propagation of radioactive substances.

Due to the different properties of a pressurized water reactor and the modular HTR-2 NPP the design basis accidents ot these plants differ substantially (see sec. 3.1). We have verified by applying the accident guideline for light water reactors analogously, if in their analysis the applicants have selected representative accidents covering the radiological consequences of similarly occurring events. The applicants have examined the following scenarios:

- Leakage of a pipe between reactor pressure vessel and primary circuit isolation valve,
- leakage of a measurement pipe containing primary coolant,
- failure of a steam generator heating tube followed by long-term failure of water separation and primary circuit pressure control.

These analyses mainly aim at demonstrating that after release of radioactive substances to the reactor building the radiation exposure in the environment will be limited sufficiently.

Further the applicants have investigated as accidents with activity discharge outside the reactor building the

- failure of the largest pipe containing primary coolant outside the reactor building accompanied by isolation of the primary circuit,
- leakage of a vessel containing contaminated water.

The reactor auxiliary building is not designed for loads from an earthquake. For this reason

- component leakages of the helium purification system and of the evaporator concentrate vessel

have been investigated by the applicants, too.

In our safety assessment we have restricted ourselves to the accidents examined by the applicants, as these events are radiologically representative for the events to be considered and cover their radiological effects. In all other events less activity will be discharged

to the environment, whereas the nuclide composition and the activity discharge mechanisms essentially remain unchanged.

We have verified the correctness of the applicants' results, the relevant input parameters, the applied calculational models and the assumed release and discharge conditions by performing independent calculations. In our safety assessment report we have described in detail deviations in our assumptions and calculations and justified them, and we have listed the nuclide-specific quantities of activity discharge. These data served as a basis to calculate the radiation exposure due to accidents.

In § 28 sec. 3 of the Radiological Protection Ordinance the dose limits in the environment of a nuclear facility are defined, which have to be observed in planning structural or technical protective measures against the radiological consequences of design basis accidents ("design basis planning limits"). In sec. 4 of the BMI guideline for assessment of the design of nuclear power plants with a pressurized water reactor according to § 28 sec. 3 of the Radiological Protection Ordinance /10/ calculational models and input data are recommended to determine the radiation exposure. These models take into account the following exposure models:

- External radiation exposure due to β-irradiation from the exhaust air (β-submersion),
- external radiation exposure due to γ-irradiation from the exhaust air (γ-submersion),
- external radiation exposure due to γ-irradiation from radionuclides deposited on the ground (γ-ground surface radiation),
- internal radiation exposure due to radionuclides inhaled by respiration (inhalation),
- internal radiation exposure due to radionuclides ingested with contaminated food (ingestion).

The calculational models and input data are not PWR-specific. Thus they can be applied unrestrictedly to activity discharges during an accident from the modular HTR-2 NPP. For this reason we have calculated the radiation exposure due to accidents based on the calculational models and radioecological parameters given in the aforementioned guideline; as a site for this plant has not yet been defined, we did not apply site-specific data. After a concrete site has been selected it has to be verified, if the applied input data still are applicable or if additional, site-specific parameters and exposure pathways have to be taken into account.

In tables 4-3 and 4-4 the results of our calculations of the radiation exposure after accidents are summarized and compared to the dose limits applicable in planning. We did not recalculate our results with respect to recent modifications of the Radiological Protection Ordinance, as these modifications are of minor influence. Our calculations confirm the magnitude of the data given by the applicants in the safety analysis report. They are distinctly below the dose limits according to § 28 sec. 3 of the Radiological Protection

Ordinance. Thus the required protective measures against the effects of accidents have been taken by the applicants according to the state of the art.

Accident	Accident doses in µSv							
	Bone	Liver	Total body	Thyroid	Kidney	Lung	Gastro-intesti-nal tract	Skin
Rupture of pipe (Ø 65 mm) Pressure relief phase	57	12	16	58	11	11	15	7.3
Core heat-up (version 1) 0-34 h not considered	0.7	0.7	0.7	120	0.7	0.7	3.4	0.5
Core heat-up (version 2) time intervals 0-34 h, 34-42 h etc.	4.1	4.0	4.0	730	4.0	4.4	24	2.8
Rupture of measuring pipe	1.0	1.0	1.0	21	1.0	1.1	2.4	1.6
Rupture steam generator heating tube, close to preheater part	1.4	0.9	0.9	90	0.8	0.9	2.8	0.6
Rupture steam generator heating tube, close to superheater part	1.7	1.2	1.1	75	1.0	1.0	2.5	0.8
Rupture of pipe in helium purification plant	2.1	7.1	7.1	9.3	7.1	7.1	7.3	2.0
Leakage of waste water evaporator	57	50	46	45	45	42	46	41
Earthquake reactor auxiliary building	240	510	490	550	490	490	500	170
Core heat-up (version 2) - Distance: 2 km -	2.4	2.4	2.4	1200	2.4	2.6	25	0.4
Dose limits § 28 sec. 3	300000	150000	50000	150000	150000	150000	150000	300000

Table 4-3: Radiation exposure in the environment due to activity discharges in radiologically relevant accidents, receiving point distance: 100 m - Adults

Accident	Accident doses in µSv							
	Bone	Liver	Total body	Thyroid	Kidney	Lung	Gastro-intesti-nal tract	Skin
Rupture of pipe (Ø 65 mm) Pressure relief phase	56	9.5	13	280	8.0	7.6	8.1	7.3
Core heat-up (version 1) 0-34 h not considered	2.2	2.6	1.6	760	1.0	1.6	0.6	0.5
Core heat-up (version 2) time intervals 0-34 h, 34-42 h etc.	14	17	9.9	4700	6.1	9.9	3.8	2.8
Rupture of measuring pipe	1.3	1.5	1.2	120	1.2	1.2	1.1	1.6
Rupture steam generator heating tube, close to preheater part	2.5	2.3	1.5	560	1.1	1.5	0.8	0.6
Rupture steam generator heating tube, close to superheater part	2.6	2.4	1.6	450	1.2	1.5	1.0	0.8
Rupture of pipe in helium purification plant	1.5	3.6	3.5	16	3.5	3.5	3.5	2.0
Leakage of waste water evaporator	57	51	45	45	44	42	48	41
Earthquake reactor auxiliary building	230	300	280	600	280	280	280	170
Core heat-up (version 2) - Distance: 2 km -	19	2.2	13	8100	5.8	13	1.5	0.4
Dose limits § 28 sec. 3	300000	150000	50000	150000	150000	150000	150000	300000

Table 4-4: Radiation exposure in the environment due to activity discharges in radiologically relevant accidents, receiving point distance: 100 m - Infants

5 Summary

TÜV Hannover e.V. has performed an independent safety assessment concerning the design of a high temperature reactor plant with two modular reactors to generate simultaneously electrical power and process steam or heat for district heating, respectively.

The assessment was based on the evaluation criteria to be applied in licensing of nuclear facilities in Germany, as e.g. the KTA-rules, and on publications representing the present state of research in high temperature reactor technology. Exceptional design features due to the new concept of the modular high temperature reactor have been dealt with in detail and their safety relevance has been assessed.

The applicants have demonstrated successfully that the radiation exposure in the environment caused by the discharge of radioactive substances from the modular HTR-2 NPP during normal operation is far below the dose limits according to § 45 of the Radiological Protection Ordinance and that the measures for radiation protection of the personnel have been planned adequately.

An important prerequisite for our assessment was to elaborate a complete and representative catalogue of design basis accidents analogously to the accident guideline for pressurized water reactors. This task inter alia facilitated to distinguish between design basis accidents and accidents beyond design basis.

Based on the design documents prepared in 1989 by the Siemens AG/Interatom GmbH project team an extensive accident analysis was performed and design requirements were developed for all components and systems. We further verified if the planned design of the buildings, systems and components as well as the operation modes of the plant will meet these requirements. These investigations were performed considering the conservative assumptions typical for licensing procedures of nuclear facilities.

A further result of the accident analysis was the identification of radiologically representative accidents and a successful verification that after these accidents the dose limits given in § 28 sec. 3 of the Radiological Protection Ordinance are not exceeded and that the necessary protective measures against the hazards of nuclear technology have been planned according to the state of the art.

Based on our safety assessment we confirm that the design of the modular HTR-2 NPP meets the safety requirements to be imposed on nuclear facilities in Germany. Our safety assessment report summarizes the design requirements as "conditions" to be met in future detailed design.

Our investigations on risk-reducing measures indicate that the modular HTR-2 NPP has pronounced inherent safety properties, which govern the plant behaviour in incidents beyond design basis.

6 Documents and Literature

/1/ Siemens/Interatom
Modular High Temperature Power Plant
Safety Analysis Report
November 1988

/2/ TÜV Hannover e.V.
Safety Assessment Report of the Modular HTR-2 Power Plant
October 1989

/3/ Nuclear Energy Act in its Version of 23 December 1959 and
Amendments of 15 June 1985 and 18 February 1986

/4/ Radiological Protection Ordinance
Amendment of 30 June 1989

/5/ Federal Ministry of the Interior
Guideline for the Assessment of Pressurized Water Reactor Design against Accidents According to § 28, Sec. 3, of the Radiological Protection Ordinance
18 October 1983

/6/ Reactor Safety Commission (RSK)
RSK-Guidelines for Pressurized Water Reactors
3rd Edition, 14 October 1981, and Annex 2 to Chapter 4.2 - Frame Specification "Basis Safety of Pressurized Components"

/7/ Federal Ministry of the Interior
Guideline for the Protection of Nuclear Power Plants against Shock Waves from Chemical Reactions by Design with Respect to Stability and Induced Oscillations and by Safety Zones
22 September 1976

/8/ Federal Ministry of the Interior
Guideline for the Radiation Protection of the Personnel during Maintenance in Light Water Reactors: The Protective Measures to be Planned in Plant Design
August 1978

/9/ Federal Ministry of the Environment
General Administrative Regulation Concerning § 45 of the Radiological Protection Ordinance: Determination of the Radiation Exposure due to Discharge of Radioactive Substances from Nuclear Plants or Facilities
21 Feruary 1990

/10/ Regulation for the Calculation of Accident Consequences According to § 28, Sec. 3, of the Radiological Protection Ordinance to Assess the Design of Nuclear Power Plants with Pressurized Water Reactors
31 December 1983