The effects of incorrect start-up and boundary conditions on the primary coolant and fuel element temperatures are checked by a so-called hot spot analysis. The hereby-resulting changes in the temperature in comparison to the nominal values are to be determined in addition to the maximum primary coolant and fuel element temperatures.

The applicant differentiates systematic and statistical faults /2.5.6–2/.

Systematic faults, which cause significant changes in the temperature, are:

- Reflector bar in the lowest position,
- Homogenizing the power density distribution in the fuel element,
- Limiting the radial number of meshes,
- Fresh fuel elements in a mesh.

Statistical faults, which can cause significant changes in the temperature, are:

- Uncertainties in the radial power density distribution
- Fault in the power measurement
- Thickness of the fuel free shell of the fuel elements.

The superimposition of the systematic and statistical faults produces a maximum fuel element central temperature of 926°C and a maximum primary coolant temperature of 866°C. The maximum fuel element temperature rises for a brief period of time to approximately 960°C during the load alternation 100%–50%–100%.

The steady-state thermodynamic analysis was performed with the THERMIX program /L 104/. It has a two dimensional non steady-state heat conducting portion for determining the energy transfer in the solid matter and a two dimensional steady-state flow part for determining the energy transfer in the fluid. The energy transfer between the solid matter and fluid is described mathematically as follows. The enthalpy absorbed from the fluid is determined from the difference between the surface temperature of the solid matter and the fluid temperature and the local heat transfer coefficients. The enthalpy used and the enthalpy lost from the fluid inside the solid matter is formed as a heat sink and a heat source. A possibly active nuclear heat source is superimposed.

The solid matter is considered homogenous. The material values are corrected according to the porosity. The local material data like for example heat conduction and heat capacity, are considered dependent generally on temperature and if necessary on the dose. A batch according to Zehner-Schlünder /L 105/ is used for the effective heat conductivity.

For calculating the temperature and flow distribution in the core, representation is done by using cylinder geometry calculation techniques; in this case the sphere fill is represented in 14 radial and 20 axial meshes. Every mesh of the sphere fill contains a part of the solid matter (homogenized fuel elements) and a part of the fluid (Helium). The pebble bed on the surface and the bottom reflector is roughly ladder-shaped.

The spent power density distribution corresponds to the representative core condition. The peak value is approximately 5 MW/m^3 in case of an average value of approximately 3 MW/m^3.

A version of the computing program COBRA-3C modified for the HTR module is used for three-dimensional comparative computation. Therefore, the reactor was divided azimuthally in 100 meshes and axially in 20 meshes by computing. The pebble beds on the surface and in the bottom reflector area as well as the thermal coupling of the sphere fill with the side reflector were not taken in to account.

For evaluating the steady state thermodynamic design we have selected following rules and regulations:

- Safety Criteria for HTR, criteria 3.1 /L 7/,
- Safety Criteria for Nuclear Power Plants, Criterion 3.1 /L 6/,
- KTA Rule 3102 /L 46/.

We derive following evaluation criterion from the above-mentioned rules and regulations:

- Satisfactorily verified computation programs are to be used,
- Systematic and statistical errors are to be taken in to account for the design,
- The guarantee of an adequate core cooling in the determined operation has to be demonstrated. In this case the fuel element temperatures are limited to approximately 1200°C on basis of the retention characteristics of the inserted fuel elements.

The requirements for the computation programs for the thermodynamic design of reactor core of gas-cooled high temperature reactors are stated in KTA Rule 3102. These requirements concern the conservation law for dimension, impulse and energy, the temperatures in the sphere fill and the thermal and hydraulic parameters.

The THERMIX program used by the applicant contains the model equations stated in the above-mentioned KTA rules for cold gas and sphere as well as the correlations between the Helium material values, heat-transfer coefficients, thermal conductivity, and the frictional pressure loss values.

THERMIX is developed specifically for the computing simulation of the thermodynamic proportions in gas-cooled pebble bed reactors in KFA. The physical model implemented in the THERMIX had proved successful in numerous experiments /2.5.6–3/. THERMIX can be regarded as basically suitable for the thermodynamic analysis of high temperature reactors as shown in these recomputations of numerous experiments.

A modified version of three-dimensional COBRA–3C computation program familiar to us was used for comparative computations with THERMIX. The version of COBRA–3C used by the applicant fulfils the requirements of the KTA Rules 3102. A proper correlation with THERMIX results can be achieved as shown with the comparative computation for two-dimensional problems /2.5.6–1/.

According to our opinion the verifications prove the suitability of the computation used for HTR safety criteria, criteria 3.1 /L 7/ programs for steady state thermodynamic design of the HTR-Module.

During thermodynamic analysis, the applicant must take into consideration all the systematic and random errors subordinated as per KTA Rule 3102. The maximum temperatures thus obtained for the fuel elements and the primary coolant show a large difference vis-à-vis the design values.

For the purpose of providing adequate cooling of the core under specified operation, the applicant must carry out computations based on representative core conditions in the following instances:

- Stationary nominal load operation,
- Nominal load with unfavorable boundary conditions for the specified operation (hot channel analyses).

These computations were made using the above-mentioned THERMIX computer program.

The highest fuel element temperature of 100%–50%–100% occurs following load alteration operation. Herein the fuel element's maximum central temperature rises to 960°C for a short time only under incorrect initial conditions and boundary conditions. Only a minor percent of the total fuel elements reach this temperature. Consequently, the maximally loaded fuel elements demonstrate a large difference at about 1200° C.

In summation, we have decisively demonstrated that, based on an assessment of the concept, no objections exist for the thermodynamic layout of the reactor core.

2.6 Nuclear Steam Generation System

2.6. 1 Functions, Construction, and Layout

The HTR two-module plant is designed as a double-block facility with two reactors, each with 200 MW thermal capacities. Therefore, two nuclear steam generating systems, each independent of the other, have been provided and share some auxiliary installations.

The function of each steam generation system is to lock in the reactor core and to transfer the thermal power generated there to the feedwater-steam cycle. Highly pure Helium is used as a coolant.

For power operation, heat transfer occurs by forced convection, wherein circulation is achieved by a primary closed-circuit blower. With the blower switched off, after-heat release from the reactor pressure vessel occurs by radiation and free convection by external cooling systems (extended surface coolers).

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

- A reactor pressure vessel with core, core elements, switch-off systems and fuel element locks,
- An intermediate pressure vessel with hot gas pipe,
- A steam generating pressure vessel and its fittings in which the primary closed circuit blower and blower butterfly valve are integrated.

The three components mentioned above are combined into one pressure vessel unit based on their function and design.

The primary gas pocket is consists of:

- The pressure vessel unit except for areas covered by the secondary coolant,
- The steam generating bundled pipes with pipe plates,
- The connecting lines on the primary side, including the shut-off fittings.

Each of the two pressure vessel units are arranged in the reactor structure in their own "primary cell". The cell for the reactor pressure vessel extends from the relative elevation by between –3 and +25 m. The steam generation vessel is offset sideways by between –12 m and +12 m. The connecting vessel to the steam generator is located in the bottom range of the reactor pressure vessel at about +3 m. The entire unit can be divided into three levels:

- The top level at about +19 m near the reactor pressure vessel,
- The middle level at about +3 m from the relative elevation of the intermediate pressure vessel,
- The bottom level at about –6 m near the steam generation vessel.

At the base of the reactor pressure vessel are connections to the blocks of fittings connected to the fuel element-charge-and discharge FCA installation and storage attachments for the spent FAB fuel elements, as well as to the conveyor system for spherical JDP switching-off elements. In the cylindrical middle region of the RDB and above the sealing spot (pressure vessel/core vessel), a DN 65 pipe line of the JEY pressure equalizer system is connected, joining the opening of the RDB with the opening of the steam generator. The pressure equalizer line has a manual shut-off fitting. From the pressure equalizer line and between the shut-off and RDM, a connection branches out to the core vessel leak test system for the core vessel JET.

The pressure equalization system opens into the connection of the KBE Helium cleaning system, whose DN 65 pipeline is connected to the top chamber of the steam generation vessel.

At the KBE connection, located at the head space of the steam generator, Helium is withdrawn out (through an intake line) into the Helium cleaning installation and fed again through the DN 65 pipe line connection on the opening of the steam generator to the primary circuit (in-feeding line). The JEG pressure relief system branches out in front of the KBE intake line, before the primary circuit switches off the connection fittings (DN 65). This line branches out to two parallel relief valve pipes.

For the primary closed circuit blower integrated at the head space of the steam generation vessel, there are three JEV oil supply connections, a forward run and a return run for the KAB nuclear intermediate cooling. Over the KAB connections, the operative cooler of the blower chamber located in the top chamber of the steam generation container is also supplied.

All the connection pipe lines are led from the pipe branches directly through the wall bushings of the primary cell. All the fittings in the connecting pipe lines are arranged outside the primary cell. This also applies for the primary closed circuit shut-off fittings in the FCA, JDP and KBE systems as well as for shut-down mechanisms of the KAB and JEV systems.

The construction of the primary circuit is a closed pressure vessel unit, appropriate to support the reactor core as safeguard for the primary gas pocket and for the transfer of thermal power to the secondary area. The ring-shaped chambers of the primary cell facilitate heat transfer by radiation and free convection from the RDB to the extended surface coolers.

The layout of the steam generating pressure vessel, offset towards the bottom and with the sideways layout of the steam generator, facilitates a very short transmission line for channeling the hot gas. In this way, there is only a negligible amount of natural circulation following interruption of the forced circulation in that the heat sink lies geodetically deeper than the heat source. As the afterheat release should be accomplished in a targeted manner only above the RDB or at the sides of the extended surface cooler, a stagnating natural circulation is advantageous.

The primary circuit components are distinctly limited by the design and layout due reactions which may be triggered from the secondary side, especially when water leaks into the primary side. The feedwater transmission and live steam transmission released from the reactor pressure vessel offset downwards through the steam generator prevents direct contact between the reactor core and the secondary circuit medium in the form of water in its liquid condition, even when increasing turbulences occur. Only an impingement with steam carrying Helium is to be presumed. Considerations pertaining to component fuel corrosion and material corrosion can, therefore, be made based on these boundary conditions.

The spatial and functional separation between the RDB and steam generator enables repair jobs on the blower, on the blower gate and on the drives under working conditions, which are radiologically largely independent of the RDB and the reactors.

Due to similar reasons and for reasons of operability, an arrangement of the manual shut-off fittings and the primary circuit fittings outside the primary cell is desirable. The pipeline lengths and connecting fittings through the walls of the primary cell up to the shut-off mechanisms, based on the above-mentioned layout, increase the risk of leakage from that primary pipe which cannot be plugged. However, this is not of much significance.

The functionally and spatially separated arrangement of the primary circuit components offers an advantage in terms of operation and, above all, operational safety. On the other hand, this pressure vessel type primary circuit unit consists of a reactor pressure vessel, a steam generation pressure vessel and an intermediate pressure vessel, which requires more materials, welded seams and a components support plan. For more information, refer to Section 2.6.2 "Pressure Vessel Unit".

The assembled components of the reactor pressure vessel are covered in Section 2.5–"Reactor Core". The assembled components of the intermediate pressure vessel hot gas line and of the steam generation pressure vessel (pipe bundle/blower) as well as support of the pressure vessel unit are covered in Sections 2.6.3 through 2.6.5 of this chapter. The process technological design and function of the nuclear steam generation system are covered in Sections 2.6.7 through 2.6.10 of this chapter.

2.6.2 Pressure Vessel Unit

2.6.2. 1 Design

The pressure vessel unit consists of the reactor pressure vessel, intermediate pressure vessel and the steam generating pressure vessel. The intermediate pressure vessel is located between the reactor pressure vessel and the steam generating pressure vessel.

The reactor pressure vessel consists of its own vessel portion and a cover, held together with 64 locking-screws. The stud bolts are in the form of expansion screws and screwed into the threaded holes of the flange ring on the bottom vessel portion.

The cover of the reactor pressure vessel consists of a lid flange and a cover cap, joined by a welded seam. The bottom end of the lid flange comes with a plate cladding mounted with the reactor pressure vessel sealing gaskets. For this purpose, two depressions are machined into the plate.

The vessel portion consists of:

- An enveloping flange ring forged to a cylindrical shell section,
- Five cylindrical, seamless shell sections,
- A bottom.

A few shell sections are designed with reinforcements to support the core assembly fittings and to support the intermediate pressure vessel. The same applies to the cylindrical shell sections of the covering flange ring on which mechanical stops, which are meant to carry loads, are located on the outer and inner sides. The top surface of the enveloping flange ring is clad with plates.

The bottom is composed of a bottom cap welded to a seamlessly forged enveloping ring. At its center a connecting piece is fixed and welded to a T-piece pipe, meant to connect the fuel element outlet and provide a breakage separation. There are also connecting pieces for the fuel element-load and the small spherical switch-off systems. A DN 500 manhole is also provided.

The reactor pressure vessel is propped up by three support brackets, weld-mounted externally onto the cylindrical cover.

The intermediate pressure vessel is seamlessly forged and welded to the connecting pieces of the reactor pressure vessel and the steam generating pressure vessel.

The steam generation vessel consists of a steam generating vessel part and a blower pressure vessel part, joined together using flange connecting stud bolts. The flange ends are clad with small plates at selected points for mounting welded lip seals.

The steam generating vessel consists of seamlessly forged cylindrical enveloping fittings, partially strengthened and welded together. The bottom closing device is made of a seamlessly forged base provided with a DN=500 manhole. This is closed with permanent blank stud bolts and a welded lip seal. The floor is welded to the bottom shell section. Based on the design of the bottom shell section, there are feedwater connecting pieces, a connecting piece for instrumentation, and mechanical stops for shock absorbers. The feedwater connecting piece is welded to a thermo-sleeve.

The shell sections have a strengthened design for mounting the steam generator's assembly fittings and connecting pieces for the intermediate pressure vessel as well as for mounting fresh steam connecting piece. The fresh steam connecting piece is, like the feedwater connecting piece, connected to a thermo-sleeve with the steam generating pressure vessel. In the top enclosure fitting, besides the fresh steam connecting piece, several connecting pieces are provided for instrumentation and for Helium control. The lifting brackets and a mechanical stop for the guide rod are placed on the cover; the connecting pieces are welded.

The blower pressure vessel part consists of the flange ring, which connects it to the steam generator pressure vessel, a covering fitting, the cover flange and the cover by a screwed connection. All parts are seamlessly forged.

The flanges are welded to the cylindrical cover. The cover is screwed to the cover flange. The cylindrical cover is equipped with different connecting pieces and ducts.

For the assessment of the pressure vessel unit described by the applicant in the /U 1/ Safety Report and the /U 2.6.2–1/ technical documents, we have taken as a basis the following evaluation criteria for assessing the constructional design:

- Safety criteria for the nuclear power station /L 6/
  - Criterion 1. 1 Principles of Safety Precautions
  - Criterion 2. 1 Quality Assurance
  - Criterion 2.2 Testing Possibility
  - Criterion 4. 1 Pressure Control Enclosure of the Reactor Coolant
- KTA Rule 3201.2 /L 48/ Components of the Primary Circuit of LWR, Part 2
- Technical Rules Pressure Vessel /L 30/
- Boiler Ordinance /L 4/
- Pressure Vessel Ordinance /L 5/

According to the rules and criteria contained in the general requirements, the design of the pressure vessel unit must be appropriate in terms of:

- Functioning,
- Stressing,
- Material,
- Manufacturing,
- Testing and serviceability, especially for repetitive tests.

The use of seamlessly forged product forms for the manufacture of the pressure vessel unit reduces the number of welded seams and basically satisfies the above-mentioned requirements. We express our opinion about the improvement of testing possibility and feasibility for repair and reconditioning.

In the assembly drawings /U 2.6.2–1/, the constructional design of the pressure vessel unit is not described in complete detail. The data on the rounding-off radius and the distance between welded seams at the wall thickness junctions are partly missing; also missing is the determination of the surface quality. In determining these data in the installation planning, other requirements must also be fulfilled with the performance of mechanized, non-destructive, repetitive tests. To a large extent, these demands cover the requirement of a design that is correct in terms of stress and manufacture. For example, to execute the connecting of the probes in the mechanized repetitive test in a U.S. test without any difficulty, the radii at curvatures should be at least 350 mm and the surface roughness must be below 20 μm. For purposes of achieving optimum connecting conditions for the probes, the waviness cannot be larger than 0.4 mm for an area of 40×40 mm.

For any repair and reconditioning measures a DN=500 manhole is always provided in the reactor pressure vessel and in the steam generating vessel. The connecting pieces' height of the manhole to be covered is larger than 250 mm. According to the conventional control regulations, the manhole must therefore be designed with a nominal width of DN=600 instead of DN=500. In the intermediate pressure vessel no openings are planned. The conventional control regulations, however, provide for this. In the installation, based on safety regulations provisos, and even according to our view, an inspection door in the intermediate pressure vessel is not required. However, it is a sound policy to provide an inspection facility even for this region.

The use of a buffered connecting piece is admissible according to the KTA Rule. In vessels comparable to light-water reactors, the model of such a connecting piece is however not the condition of the art of science and technology. The repetitive test for the buffer is not possible with the mechanical testing attachments because of the studs, which connect the compressed glass bushing and blank cover of manholes with the buffered connecting pieces. The connecting pieces of the compressed glass bushings and the manholes should therefore be designed as welded connecting pieces, as portrayed, for example, in document /U 2.6.2–1/.

The basic requirements of conventional and nuclear technological control installations, especially KTA Rule 3201.2, with regard to the design of the pressure vessel unit, can be fulfilled by taking note of the mentioned proposals, and the requirements issued in the BMT criteria can be fulfilled.

2.6.2.2 Strength

As described in the Safety Report, the strength test is done in two steps. In the first step the dimensions of the load carrying components are determined, wherein temperature, pressure and other loads are considered. In the second step the exact strength test is produced by application of detailed stress analyses. In these analyses, the details are derived from the geometry of the component, component temperatures and all the loads to be considered and also their frequency.

On the basis of the proposed materials and the intervening pressure and temperature stresses, the strength-based layout of the pressure vessel unit is to be basically prepared by applying the KTA rules established for light-water reactors. With regard to the exception for the fresh steam connecting piece stressed at 540°C, it will be separately discussed.

Stresses

Data are compiled in the /U 2.6.2–4/ document on the dimensioning to be done for the stresses subordinated in the /U 2.6.2–2/ reactor pressure vessel, /U 2.6.2–3/ steam generation pressure vessel and /U 2.6.2–4/ intermediate pressure vessel. These reports contain the data pertaining to:

- Design pressures,
- Design temperatures,
- External mechanical loads to be covered.

We have checked the reports presented to us regarding the stresses and we confirm that they are within adequate conservative limits for the dimensioning of the components. However, they are of provisional nature as their sizes are to be determined from the accurate system data, component sizes, spatial layouts and their specific requirements from the EVA and EVI events.

The loads to be taken into consideration in the stress analysis are given in report /U2.6.2–5/. The loads are classified therein according to the following types of load:

- Normal operation,
- Anomalous operation,

- Testing incidents,
- Anomalous incidents (emergency and damage events).

The stress stages A, B, P, C, and D according to KTA Rule 3201.2 are assigned to these types of load.

We consider the selected design loads for the design of components of the pressure vessel unit adequate. Changes may occur when final seismic data are available for a specific site. However, these values do not have any influence for assessing the construction design of the pressure vessel unit.

Document /U 2.6.2–5/ contains the types of load to be considered according to KTA Rule 3201.2 and arranges them in a suitable manner according to the stress stages, according to which the admissible stress limits according to the KTA Rule 3201.2 are to be determined. The individual stress dimensions are to be taken as specifications during the planning stage of the installation construction. The basic building potential of the components exists for the types of load available.

Stress Limitation in the Components

The objective of the strength calculations is to provide proof that the stresses (tensions and expansions) in the construction components are limited in such a manner that a failure during their lifetime can be excluded. For the component ranges in which the design temperature is below 350°C, the admissible stress limits are fixed according to he KTA Rule 3201.2, wherein the primary and secondary stresses as well as peak stresses are limited. The acceptability of the occurrence of load type is evaluated by special fatigue analysis.

For fresh steam connecting pieces, operating temperatures of 540°C occur. At these temperatures, the time response of the material-characteristic values is to be considered. Therefore, in addition to the short-time characteristic values of materials, even the creep strength of the materials is to be considered in the design. This is done by complying with the control regulations demands of conventional engineering, such as those given in TRD and AD pamphlets. For advanced analyses of the mechanical behavior, the ASME Code, Section III, Code Case N47 /L 31/ is consulted, based on which a decision can be taken about the creep phenomena and creep strength damages.

We consider as admissible the application of KTA Rule 3201.2 for fixing the admissible tensions in the pressure vessel unit because the application limits and material provisions of this rule are to be followed. For dimensioning and analysis of the mechanical behavior of the fresh steam connecting piece, the German body of regulation TRD /L 30/ or the ASME Code Section III, Code Case N47 can be consulted. Both rules contain the basics, instructions and requirements for the safe design of construction components keeping under consideration the creep behavior noticed at time of construction. The unusual problems which are encountered during welding of materials of different properties are to be considered here. The demands pertaining to the scrapping of breakages are handled in Section 2.6.2.3.

Design calculations

To prove that the construction components of the pressure vessel unit can be adequately dimensioned, the applicant has carried out design calculations on the basis of the design loads and provisional assumptions of load. These design loads and provisional design loads are compiled in report /U 2.6.2–2/ for RDB, in report /U 2.6.2–3/ for the steam generator and in report /U 2.6.2–4/ for the intermediate pressure vessel.

The design pressure of 70 bar covers, in this context, the selected operating pressure of 60 bar of the installation. The design temperature of 350°C is selected by covering all operative and anomalous incidences with the /U 2.6.2–8/ internal pressure stress. A minor instance of exceeding of the design temperature at 355°C occurs in the pressure relief load type. The provisional loads, connecting pieces and supports of the pressure vessel unit to be used in the design are selected according to /U 2.6.2–9/ in such a manner that they are larger than the forces obtained from the earthquake stresses and aircraft crash stresses.

For all areas of the pressure vessel unit /U 2.6.2–10, U 2.6.2–11 and U 2.6.2–12/, the design calculations were carried out wherein it was proved by the manufacturer that the primary stresses for the above-mentioned design data conform to the admissible limits according to KTA Rule 3201.2. Besides the internal pressure stresses, the external loads stresses are applied for connecting pieces and load-lifting brackets /U 2.6.2–13 and U 2.6.2–14/.

We have checked the design calculations in the light of completeness and correct application of the set of rules we hereby confirm that the requirements of the set of rules can generally be complied with.

Problems, which bring into question the adequate dimensioning that, were not firmly established. The dimensioning of the fresh steam connecting piece is done by complying with the stress according to TRD or ASME Code, wherein the creep behavior of the materials used at 540°C was taken into consideration. Far-reaching creep investigations are to be executed at the time of planning the construction within the framework of mechanical behavior analysis.

Mechanical Behavior Analysis

Within the framework of the design phase, no analysis of the mechanical behavior according to KTA Rule 3201.2 was carried out by the applicant. However, to evaluate the mutual influence of temperature, pressure and obstructed thermal expansion on different vessel units, we have designed the entire pressure vessel unit as a finite element model and calculated the various load types (Figure 2.6–1).

On the basis of our analyses and from our experience with stress analyses of different components, we come to the conclusion that the analyses of the mechanical behavior can be carried out according to the KTA Rule and that the required stress limits can be maintained.

Demonstration for Brittle Material Fracture Failure Safety

KTA Rule 3201.2 requires, within the framework of mechanical behavior analysis, that the safety against brittle material fracture failure is demonstrated. This requirement of basic safety is met by fulfilling the viscosity concept according to RSK guideline Section 4.1.2. Pressure tests of components reveal an additional protection against brittle material fracture failure.

In the case of components that suffer a change during operation, additional demonstrations are required. In this context, for RDB in which the viscosity decreases with neutron radiation, calculations were made on the basis of Rupture Mechanics /U 2.6.2–15/. They show that by maintaining the specified pressure and temperature conditions during start-up and shut-down as well as during pressure tests, under the conservative assumption of an error resulting from a depth of ¼ wall thickness, an adequate safety against brittle material fracture failure of RDB exists. The radiation embrittlement was considered with an estimated displacement of brittle material fracture failure transition temperature of 10°K. The possible radiation embrittlement is verified by an irradiation program /U 2.6.2–16/ and monitored through the accompanying live suspended probes /U 2.6.2–17/ (see here Section 2.6.2.4).

> Proceeding from the presented calculated proofs and by taking into consideration the radiation embrittlement monitoring program, it can be ascertained for a reactor pressure vessel that the requirements of the KTA Rule for the brittle material fracture failure demonstration can be complied with.

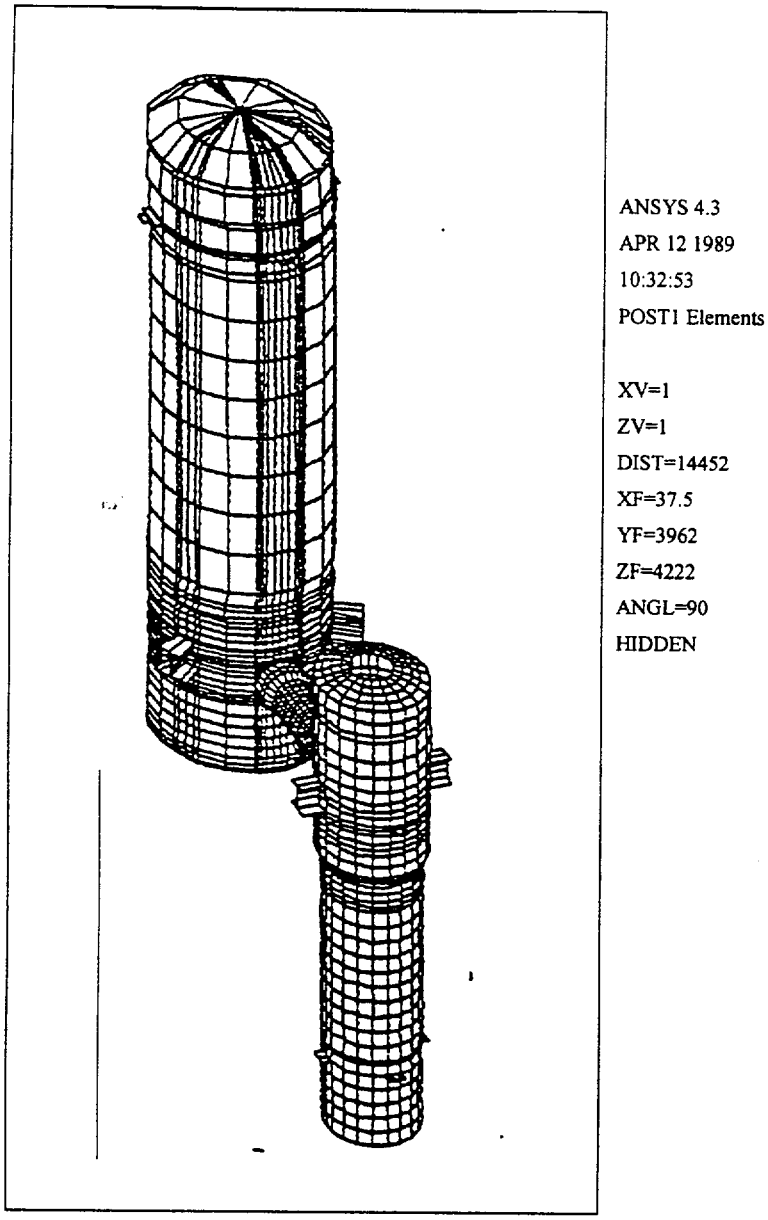

Figure 2.6.1: HTR pressure vessel unit (RDB+DE+connecting pressure vessel) under internal pressure

2.6.2. 3 Rupture Assumptions

The pressure vessel unit is designed for the presumed failure of the largest connected pipeline (DN 65). Larger leakages are not presumed. The 2F-rupture of a steam generator heating pipe is assumed to be leak covering the primary side and the secondary side. This requirement, determining the design concept of the HTR module facility, is based on the premise that because of the planned measures, a failure of the pressure vessel unit can be excluded due to leakage or rupture during the lifespan of the installation. These measures lead to:

- Optimal constructional design
- Conservative design
- Suitable material selection and high material quality,
- High quality during manufacture and testing.

Especially the

- High cost,
- Limitation of stresses by adhering to the KTA regulations and
- Reliable determination of the possible error conditions in the construction components after production and the operating cycles

justify, in the view of the applicant, the exclusion of cracks, which can penetrate through the walls of the vessel. Consequently, ruptures or leakages in all the wall regions of the pressure vessel unit are excluded. As a further substantiation of the exclusion of ruptures, failure mechanics analyses were carried out by the applicant at the following spots of the pressure vessel unit /U 2.6.2–18/:

- Connections of the connecting piece to the RDB (fuel element outlet pipe, connecting pieces of the small sphere shut-down system, fuel element-feed pipe),
- Contact pressure container,
- Steam generator fresh steam connecting piece,
- Steam generator feedwater connecting piece.

The proofs obtained from failure mechanics are based on the following basic boundary conditions:

- Material Ductility

The ductility properties of the materials in the temperature range below 350°C are known and they are guaranteed by using basic safety concepts /L 12/ for all operative conditions at the required level. For the areas with operating temperatures above 350°C (fresh steam temperature), the requirements of the KTA Control Regulations 3201.1 can be basically fulfilled based on materials verification. Hence the calculations based on failure mechanics presume that the ductile material characteristics in the calculations of critical lengths of cracks are both in the lower and in the higher temperature range.

- Hypothetical Errors

Proceeding from the results of the non-destructive tests during production and of the repetitive tests, the extent of error detectable with certainty is defined as errors that can be presumed as the maximum.

– Stresses

The stresses resulting from the operative instances and anomalous incidents are utilized. In the crack growth analysis, the number of cycles to be expected and, additionally even the stress duration on the fresh steam connecting piece, are utilized for considering the creep fatigue.

The applicant proves with the help of failure mechanics calculations that:

- The DN 65-leak design is selected, covering the entire pressure vessel unit,
- The 2F-hot pipe leak covers the possible leakages between the primary side and the secondary side,
- The magnitude of leaks, as they were ascertained on the basis of failure mechanics, can be promptly detected by leakage determination.

In the following paragraphs, the individual proofs of failure mechanics are treated.

Fuel Element Outlet Pipe, Fuel Element Feed Pipe and Connecting Piece for the Small Sphere Shut-down System (KLAK)

The calculations based on failure mechanics are given in the document /U 2.6.2–19/. The stress calculation takes into consideration the internal pressure and external forces in anomalous incidents for the computation of the /L 31/ critical crack sizes. For the determination of the crack growth, the internal pressure stresses and thermal stresses at the time of start-up were superimposed. By doing so, according to the applicant's details the other operative stress conditions are dealt with on a conservative basis.

The scope of the error on the inner surface is defined as an initial error in the range of the above-mentioned error and this is presumed to be the maximum. A comparison of this error with that of the calculated critical error magnitude indicates a high level of safety against failure of critical components. The crack growths for the hypothetical error were calculated for 500 load cycles (specified for the reactor service life) and it is negligibly small.

Based on the assumption that after many arbitrary load cycles, the crack would penetrate through the wall of the pipe only after multiple times the service life of the reactor, wherein its crack length compared to the start-up length is considerably increased. This increased crack length was used to determine the surface area of the leak with a presumed crack under the operative conditions. The results of the calculations of failure mechanics are compiled in the following table:

				Start-up error at the crack			
	Inner radius (mm)	Wall thickness (mm)	Critical crack length (mm)	Length (mm)	Depth (mm)	Error	Area (mm)
BE outlet pipe	600	85	2180	25	4.2	330	25
BE feed pipe/KLAK	175	25	460	25	4.2	144	5.8

The approach adopted by the applicant for the proof of rupture exclusion is approved by us. The calculation of critical error magnitude is done with the KWU program DLECK by taking into consideration the ductile material characteristics. Circumferential cracks were considered which, according to the RSK guideline "Framework Specification for Basic Safety Errors " /L 12/, are to be presumed only for welded seams.

Comparative computations for checking the program results were carried out by us by taking into consideration the studies of Eibner / U 2.6.2–20 /, U 2.6.2–21/ and the obtained results were positive. The ascertained critical crack lengths are very large on account of the low stresses even under the consideration of anomalous incident loads and they lie considerably outside the conceivable error magnitudes. According to our assessment, the error measurements applied for the crack growth computation can be found out with the ZfP method. The resulting cracks lengthening for the operative load alterations within the serviceable life of the installation (500 times start-up and shut-down, approximately 3000 times partial load alterations) were determined by us by using ASME Code /L 32/ based on the given methods and material values. The results were negligibly small.

In addition, the possible crack characteristics determined by assuming an arbitrary number of load cycles was examined by the applicant and also by us. In the relevant literature /L 109/, it is reported about the results of investigations on cracks lengthening up to the crack (leakage) under tensile and bending stresses. Accordingly, the maximum possible crack length is limited to about three times the wall thickness under tensile stress only; while under bending stress only (*e.g.*, in the case of thermal stresses) the crack lengths are approximately 18–20 times larger than the wall thickness. For the BE outlet pipe, the applicant has computed, with conservatively determined stress values (tensile stress 25 N/mm^2 and bending stress 35 N/mm^2), a crack length of about four times the wall thickness; while the computations according to /L 109/ gives about double the value (8.6 times the wall thickness).

The stress distributions determined more accurately by us for the BE outlet pipe presents a more favorable ratio of the bending stress to the tensile stress and, consequently, lead to a possible crack length of about seven times the wall thickness. The leakage surface area obtained thereby under operative stress is approximately 130 mm^2.

In the case of the feed pipe, we confirm the results of failure mechanics obtained by the applicant because the stress distributions contain only a small component of bending stress so that the crack lengthening associated with it is not developed.

In summary, we come to the conclusion that a guarantee is demonstrated for the exclusion of ruptures in the above-mentioned component. A leakage surface area that can be excluded during the lifespan of the installation, determined on the basis of stress duration, is lesser than that of the rupture surface area of a connecting pipe DN 65 based design. The length of crack obtained thereby lies distinctly below the critical crack length so that a critical failure can be excluded.

Intermediate Pressure Vessel

The applicant employs the stresses mentioned in /U 2.6.2–22/ for the computations of failure mechanics as a basis for excluding the ruptures in the connecting pressure vessel /U 2.6.2–32/. Accordingly the following are determined:

- Membrane stresses caused by internal pressure,
- Membrane stresses caused by anomalous incidents (earthquakes) and frictional force,
- Bending stresses due to earthquakes, and
- Bending stresses due to the blocking of shock brakes

These have been taken to act simultaneously.

This superimposition of the stresses is admissible as the bending stress is caused by an external factor and results in a fairly uniform distribution over the pipe wall. The bending stress is conservatively raised still higher by a factor of 1.5 for the determination of a possible initial error of the critical crack length and of the crack growth. The calculated critical crack length is adequately large compared to a possible, assumed initial error or the maximum possible error in the event of a crack after several load cycles (70 reactor lifespans). The maximum possible leakage area has been determined with this maximum possible error. The data and results of the computation based on failure mechanics are compiled in the following table.

				Start-up error at the crack			
	Inner radius (mm)	Wall thickness (mm)	Critical crack length (mm)	Length (mm)	Depth (mm)	Error	Area (mm)
Intermediate pressure vessel	750	50	370	25	4.2	150	100

An important viewpoint in the assessment of rupture exclusion in the intermediate pressure vessel is governed by the fact that the resulting stress conditions can be reliably demonstrated. With the operational loads due to internal pressure and operational axial forces (from the frictional component), result in only negligible stresses which lead to high level of safety against failure due to cracks.

The failure of the shock brakes in the steam generator was examined as an exemplary anomalous incident. In our comparative computations, the load type "blocking of the shock brake" leads to bending stresses of about 125 N/mm^2 using the FE-method in which the value determined by the applicant using simplified method /U 2.6.2–23/ is confirmed. The stresses from the load type "the failure of the shock brake during earthquake" are considerably above those that are established for the above-mentioned "Blocking of the shock brake" as a consequence of the thereby caused thermal expansion that was obstructed.

The examination of earthquake stresses reveal a large bandwidth of stresses, which can be established for the earthquakes load types. Thus, the bending stresses in the intermediate pressure vessel during an earthquake are about 200 N/mm^2. The safety against crack failures is still always guaranteed, as with these stresses and the additional membrane stress of 280 N/mm^2, the error to be presumed in the intermediate pressure vessel is still basically lesser than the critical error magnitudes.

However an increase in stresses would considerably reduce the safety against crack failures. To limit the stress during an earthquake, we recommend having a deflection limiting device on the steam generator shock brakes because thereby the stresses can be limited to the value of the load type "blocking". In the event of such a construction, it is to be tested whether it is possible to completely do away with the shock absorbers.

We confirm the error magnitudes calculated by the applicant using our comparative computations according to /L 107, L 108/. Our comparative computations with respect to the crack surface areas and crack growth have yielded values, which are only negligibly above those of the applicant. Consequently, the rupture exclusion for the intermediate pressure vessel is accounted for by the computations based on failure mechanics. Leakages as a result of cracks on the walls, which could indeed be excluded during the lifespan of the installation, have a magnitude of about 1 cm^2 and they are covered based on the assumption of a comprehensive appraisal of a pipeline with DN 65.

Feedwater and Fresh Steam Connecting Pieces

The approach for establishing the ground for the rupture exclusion in both the above-mentioned connecting pieces is described in the /U 2.6.2–24 and U 2.6.2–25/ documents, and they are comparable with the approach for the other ranges of the pressure vessel unit.

Based on design loads/U 2.6.2–26, U 2.6.2–33/, the stresses caused by operative and anomalous incidents (earthquakes, pipe rupture) are considered in the stress computations. The critical crack lengths are calculated from the maximum occurring stress conditions of different circumferential welds, by taking into consideration the different material properties.

In the case of feedwater connecting piece, the 20MnMoNi55 material is used for connecting pieces because of the design temperature of 350°C. The connecting piece between the pipe plate and feedwater line is made of the Incoloy 800 material.

The material X20CrMoV121 is chosen for the fresh steam connecting piece because of the operating temperature of 530°C; that with the pressure vessel wall over an intermediate piece is made of X10CrMo9 10 and connected with the pipe plate over an intermediate pipe made of Incoloy 800.

The computations with regard to the critical error lengths were done by the applicant with the axial forces mentioned in the /U 2.6.2–24 / and /U 2.6.2–25/ reports on the inner side of the connecting pieces.

For purposes of determining the crack growth of an assumed error of 4.2 mm depth and 25 mm length, the applicant takes into consideration 500 stress cycles, arising from the internal pressure load which are obtained from the start-up and shut-down cycles. In the case of fresh steam connecting piece, additionally the lengthening of the crack due to the material creep is calculated and the crack growth from stress cycles is added. Under these mentioned boundary conditions, the investigations yield, for the planned reactor life duration, a negligible crack growth in the depth direction and in the longitudinal direction.

Because in the computations only membrane stresses, but no bending stresses, were considered, the assumed initial error develops after several random load cycles into a stable crack whose length remains limited to three times the wall thickness. This error length is rather small compared to the critical crack lengths and results in a leakage surface area towards the secondary side which is smaller than the rupture surface area of a DE heating pipe, and falls below the rupture surface area of the largest connection line (DN 65) towards the exterior.

On the basis of these results, the applicant comes to the conclusion that the leak-before-rupture-behavior is proved for all instances of intervening load, and the maximum, conceivable leak surfaces – which are indeed excluded during the life of the reactor – are smaller than the design values.

We accept the method of proof employed by the applicant for rupture exclusion. The extension of the rupture mechanics method to the high temperature areas is also approved by us because they correspond to the present-day status of such proofs /L 110/. However, we have established that the proofs of the applicant on many points do not adequately take into consideration the given material properties and stress cycles. This is explained in the following paragraphs.

The / design loads U 2.6.2–26/ for the calculation of critical crack lengths have been revised on the basis of our tests, according to which the reports /U 2.6.2–24, U 2.6.2–25/ had already been prepared. A considerable rise on the axial force was obtained, resulting from ruptures of the subsequent pipe lengths (fresh steam line and feedwater line) on the inner side of the connecting pieces. Consequently, even the stresses in the Incoloy-ferrite welded connection region increase so that the critical crack lengths are reduced.

In the stress computations, the thermal stresses from start-up and shut-down cycles were not taken into consideration. Furthermore, stresses from the obstructed thermal expansion are missing; these stresses develop by the diverse temperature expansion behavior of the Incoloy and X20CrMoV121 connected materials in the fresh steam connecting piece as well as Incoloy and 20MnMoNi55 on the feedwater connecting piece. The finite element analyses have resulted in additional bending stresses of about 180 N/mm^2 in the vicinity of the fresh steam connecting pieces.

At the time of start-up and shut-down additional bending stresses are generated in the walls (75 mm wall thickness). All power changes in the installation are also linked with changes in the fresh steam temperature; the other small stress cycles result in a higher frequency. Additional thermal stresses of over 250 n/mm^2 occur, which according to our assessment cause damages through load alternation and creep fatigue, if the internal part of the fresh steam connecting piece is not protected from outside by heat insulation against cold gas.

The applicant's computations pertaining to the crack progression take into consideration only the stresses arising from operative internal pressure. Hence, the applicant obtains negligible crack propagation for the Incoloy area and very low but unobjectionable values for the ferritic area.

The additional thermal stresses occur in the Incoloy area and in the immediate connecting ferritic region (X20CrMo121 in the fresh steam lines and 20 MnMoNi55 in the feedwater lines). The increased stresses (bending stresses), from the obstructed thermal expansions as a consequence of different material behavior and from the start-up and shut-down cycles, in our opinion, lead to a increased but unobjectionable crack growth from cyclic loads in the fresh steam connecting pieces and in the feedwater connecting pieces.

Due to the increased operating temperature of 530°C in the fresh steam connecting piece, additionally such a high creep crack growth arises in the fresh steam connecting piece in the depth direction so that an assumed initial error of 4.2 mm depth and 25 mm length penetrates through the pipe wall thickness of 75 mm within the stipulated operating time of the reactor installation. Due to the very high bending stress component, one has to also reckon with a very large crack growth in the longitudinal direction of the error, which can lead to large crack surface areas right up to the circumferential tearing in the region of the Incoloy/X20CrMoV121 composite seam.

In 1987, the Minerva component test /L 111/ was started based on the experience gathered with HTR components for this component range, and because a rupture exclusion was recognized as critical; this test was to provide proof for a safe operation during the lifespan of the Incoloy/ferrite welded seam joint. In this context, the above conditioning indicates that near the welded seam, errors were not located through non-destructive testing, and these errors could lead to crack growth by cyclic stresses or creep stresses. The initial results of this project are reported in /L 117/. Accordingly, very high thermal stresses (about 200 N/mm^2) with peak stresses of up to 400 N/mm^2 are obtained whose level can be brought down by optimizing the design of the welded seam structure.

Even within a short period of application during the creep test, large expansions are obtained which occur initially in the buffer layer and later in the heat affected zone (WEZ) of X20CrMoV121. The creep strength in the mixed joint is considerably lesser than in the basic material. After about ⅔ of the planned test duration, massive creep damages occurred in the ferrite WEZ region. In addition, cracks were observed in the fusion line.

The report comes to the conclusion that in the region of 550°C crack growth was observed; that could lead to a considerable reduction of the load-carrying cross-section and consequently the life could be considerably reduced due to the creep stress. The test report does not end on a positive note regarding the expected operational safety of the composite seam joint, but refers to the additional information to be obtained from the project and possible optimizations.

On the basis of the results obtained from our computations on the crack growth, we come to the conclusion that the design of the fresh steam-connecting piece of the steam generator placed before us, does not guarantee rupture exclusion in the inner region. There exists the possibility to:

- Decrease the thermal stress level by further optimizing the composite seam joint,
- Obtain a lower creep damage from the Minerva tests with reduced external stresses
- Prove an effective reduction in the crack growth due to creep through far-reaching, detailed computations of stresses by taking into consideration the relaxation behavior.

Alternatively, we have suggested to the applicant to undertake a modification of the design.

The applicant has presented us with a design change in the fresh steam connecting piece which avoids the usage of a composite welded seam joint of Incoloy 800 and ferritic material in the rupture exclusion regions and design temperatures of above 350°C /U 2.6.2–34/.

After that, the entire fresh steam connecting piece up to the connection to the connecting pieces in the pressure vessel shell and up to the welded seam joint of the fresh steam line were produced from Incoloy 800. The composite seam joint between Incoloy 800 and 20 MnMoNi 55, the material of the steam generator pressure vessel connecting piece, still only attains temperatures below 350°C in the changed position and consequently, it is not stressed up to creep strength anymore. The stresses created by different thermal expansions in this seam region have only a small influence on the widening of cracks during cyclic load because the number of start-up and shut-down cycles brings about a negligible crack growth during the reactor life.

For the computations of the Incoloy 800-section of the fresh steam connecting piece, the characteristic material values of the draft standard of DIN 17460 (status 11/88) were made use of. This DIN standard evaluates the results of research projects on high temperature reactor with regard to material characteristics, material composition and processing, and incorporates Incoloy 800 with material number 1.4958 under the material designation X5 NiCrAlTi 3120 in which a normal and a re-crystallization annealed form of supply is possible /L 157/. Because the temperature range of 500° to 570°C lies higher than the strength values of the material in the re-crystallized form, this was selected for the fresh steam-connecting piece.

For the computations of rupture mechanics the following load types which cover the planned mode of /U 2.6.2–25/ operation were selected by the applicant. In doing so, the stresses from internal pressure, external moments and thermal stress were demonstrated.

Load type	Stage	Number	Creep time / h /
Normal operation	A	–	2.8×10^5
Start-up and shut-down	A	400 each	160 each
Load Ramps	A	50,000	≤555
Load Jumps	A	20,000	≤140
Restart-up after RESA, Preparation for hot start or shut-down	B	400	40
Incorrect closure of a FD fitting	B	20	10
Blower run-up through control error	B	10	100
Feedwater throughput run-up through control error	B	10	0.07
Pressure release during anomalous incident	C	3	30
Pressure testing on the secondary side	P	5	

For the determination of temperature stresses the applicant has carried out stress analyses for the following load types:

- Shut-down

  Here, the steam temperature gets reduced quickly. The pipe wall is cooled on the steam side and is subjected to tensile stresses, which again get reduced due to the temperature equalization. The transient stress leads to crack growths through cyclic loading and to creep crack growths as long as the temperature of the pipe wall is above 500°C.

- Initial operation according to RESA
  Here, hot steam with a temperature of 620°C is led through the connecting piece for a short time.

- Acceleration of the feedwater throughput
  The steam temperature is reduced to about 360°C by the increase in the feedwater throughput.

These load types generate considerable thermal stresses in the pipes, which, in addition to the stresses from mechanical loads, are considered during the computation of the crack progress.

For proving the rupture exclusion in the fresh steam connecting pieces, examination is made of the crack characteristics for all other welding seams, which are representative of circumferential seam, by rupture mechanics computations for an assumed initial error of depth a=4.2 mm and length l=25 mm /U 2.6.2–36 /. To this end, the crack growth due to cyclic load and material creep is calculated, wherein a creep for those stress conditions is considered in which the temperature lies above 500°C. Because the increased temperatures and stresses emanating from the stationary cycles crop out for only a short time, they bring about only a small crack growth.

The crack growth for the above-mentioned operating conditions show, for the assumed initial crack, an elongation of 0.05 mm in the direction of depth and 0.02 in the longitudinal direction during the reactor's life. Because the smallest critical penetration crack length is 780 mm, the exclusion of rupture of the fresh steam connecting piece is always guaranteed during the specified duration of operation. Advanced computations of the elongation of cracks over and beyond the specified life of the installation show that leak sets in after about 400 lifespans of the installation. The pertinent crack length at 600 mm is still below the critical crack length, so that a stable leak is to be expected. The possible leak surface area is equal to approximately 115 mm^2 and it is smaller than the leak obtained from the rupture of a heating pipe; this has been taken into consideration for the installation design.

The design measures undertaken by the applicant are suitable for minimizing the stresses of components of fresh steam connecting pieces, wherein especially the stresses from the obstructed thermal expansion of dissimilar materials at temperatures over 350°C are avoided. The stress computations carried out by the applicant cover the most important operating conditions corresponding to the requirements, as they are necessary in their application to elicit verification based on rupture-mechanics. We have checked a few computations concerning a few load types through comparative computations and we confirm the results of the applicant.

We have evaluated the computations done for the determination of the cracks progress on the basis of literature /L 158/ and on the basis of our own computations. Based on that, we can confirm that for the planned mode of operation and for the specified lifespan, leakages and ruptures can be excluded because even the assumed large inherent errors increase ever so slowly in the depth and longitudinal direction that there from no crack penetration through the wall can develop.

More detailed considerations of the cracks progress shows that only after operating times that exceed the lifespan of the installation by about 400 times, a leakage formation is possible. This, however, does not lead to a rupture of the fresh steam connecting piece. The leakage surface areas that are possible to such a maximum extent but, which are generally not to be expected in the life time of the installation, are determined sufficiently conservatively, and their effect is covered by the designed leakage surface area (rupture of a steam generator heating pipe).

2.6.2.4 Materials

Heat-resistant, fine grained, heat-treated steel 20 MnMoNi 55 (material: 1.6310) is used for the manufacture of the pressure-carrying walls of the pressure vessel unit. The seamless, hot-pressed or forged spherical caps, rings, flange and connecting piece are welded together to the pressure vessel unit or they are mutually connected with stud bolts. During the manufacture of the product forms for the reactor pressure vessel, the material is melted in an electrical furnace and vacuum cast for obtaining special purity and uniformity. The feedwater entry connecting piece and fresh steam exit connecting piece of the steam generator pressure vessel are also manufactured from the 20 MnMoNi 55 material. The material is certified for application in nuclear engineering installations and for application up to 375° C, and suitable for the specified field of application according to the applicant.

The fresh steam generated in the steam generator pressure vessel leaves the latter at a temperature of about 530°C. Materials, which can be applied simultaneously with this temperature, and other loads (internal pressure, forces, moments), are subjected to a creep stress. Due to this reason, the heat resistant materials 10 CrMo 9 10 (1.7380), X 20 CrMoV 12, (1.4922) and X NiCrAlTi 32 20 (Incoloy 800, 1.4876) are used for the creep stressed regions of the transition piece (thermo-sleeve) and fresh steam line.

The applicant describes the present-day status of qualifying materials and welded joints (among others even composite joints) which are intended for application in the high temperature region /U 2.6.2–27/. Herein, the creep strength characteristics, processing and rupture mechanics based evaluation (viscosity characteristics) are discussed in detail-

The stud bolts of the reactor pressure vessel are made of 26 NiCrMo 14 6 material; those of the steam generator pressure vessel and blower pressure vessel are made of 20 NiCrMo 145/II material. The 34 CrNiMo 6 material is employed for the nuts of the stud bolts of both the components.

Welded parts on the pressure carrying walls of the pressure vessel unit are, as a rule, made of the same material as the pressure carrying walls themselves.

For assessing the suitability and eligibility of the materials for the pressure vessel unit (basic materials, welding filler metals), the evaluation criteria as mentioned below are used:

- Safety Criteria for Nuclear Power Stations /L 6/
  Criteria 1.1, 2.1, 2.2, 4.1

- KTA Rule 3201 /L 48/
  Components of the Primary Circuit of LWR
  Part 1: Materials
  Part 3: Manufacture

- AD Material Standards WO /L 28/
- AD Materials Standards WO /L 29/
- Technical rules for pressure vessel /L 30/
- Steam boiler ordinance /L 4/
- Pressure vessel ordinance /L 5/

A so-called instant program and the "component safety" research program (FKS) was undertaken as a suitability test of the 20 MnMoNi 5 5 steel. The results of the material appraisals, by specialists, contained in VdTÜV Material Data Sheets /L 163/ and those in the KTA Rule 3201.1 /L 48/ were considered.

To safeguard material data relevant for the HTR module reactor pressure vessel, additional investigations are necessary with regard to the influence of neutron radiation in the temperature range below 250°C. These investigations are to be carried out in the framework of a progressive and time accelerating /2.6.2–16/ irradiation program. Therein, the relevant material data depending on the neutron influence is to be determined for the HTR operating temperatures. In addition, proof has to be furnished through comparative irradiation with lower neutron flux density so that the time-accelerating irradiation with high neutron flux density brings out conservative results. Through a re-examination of the specimens, the shift in the transition temperature of ΔT_0=10 K should be accounted for based on the rupture mechanics computations. Should it result in a higher value here, then the irradiation of additional rupture mechanics specimens is specified.

For testing the radiation embrittlement during the operation of the HTR module, specimens from the original materials of RDB are irradiated in the framework of an accompanying irradiation program in the HTR module /2.6.2–17/. Because this irradiation cannot be done with a leading power factor due to installation-specific reasons, the sampling was upgraded to meet the requirements of the KTA Rule 3203, with a view to evaluating with certainty the possible increase in the radiation embrittlement for the period up to the next sampling. By locating the sample storehouse directly above the RDB wall, it can be ascertained that representative temperatures and irradiation fluences are present.

From our point of view, the 20 MnMoNi 5 5 material is suitable for the provided application by complying with the requirements of the above mentioned program, the VdTÜV material standards and KTA rule.

For proving the usability of materials, 10 CrMo 9 10, X 20 CrMoV 12 1 and X 10 NiCrAlTi 32 20, in the nuclear facilities, in addition to the conventional certification, extensive additional investigations were carried out to recognize the primary suitability of the materials for the specified application range. A few investigation programs have still not been completed /U 2.6.2–27 /. We have delved into the special problems of the composite joint of X 20 CrMo 12 1 and X 10 NiCrAlTi 32 20 for assessing the rupture assumptions made in the previous section. The constructional changes introduced in the fresh steam connecting piece region, has considerably simplified the problem of this welded joint.

The welded seams lie at present on the one hand in temperature ranges below 350°C so that here no special requirements arise from the material creep, and on the other hand in regions for which an exclusion of rupture need not be proved. Consequently, the requirements of conventional set of rules can be applied here.

The use of 26 NiCrMo 14 6 material for the stud bolts and 34 CrNiMo 6 material for the nuts has been tested for the reactor pressure vessels in light-water reactors. Their suitability has been proved and we have no objections against their usage.

2.6.2.5 Periodical Tests

The applicant has described the concept of periodical tests on the pressure vessel unit in the Safety Report and in other documents /U 2.6.2–28, U 2.6.2–31/. Accordingly the following test procedures are to be applied:

- Ultra sound test procedures for checking the material contents and material sizes near the surface,
- Eddy current testing for checking material sizes near the surface,
- Magnetic powder testing method and penetration method of testing as well as visual inspection to check the material surfaces,
- Pressure test for the integral proof of integrity and operational safety.

Basically, ultrasound testing is done on pressure-retaining weld seams, on welding seams at the pressure-retaining wall, on connecting seams and on highly stressed basic material regions /U 2.6.2–29/. As a rule, remote-controlled, mechanized testing attachments are used for this purpose. The applicant, however, presumes the possibility of even manually carrying out these tests in regions of low radiation.

Eddy current testing is employed for testing the near the surface regions of stud bolts, nuts and threaded blind holes of the reactor pressure vessel. Basically, the tests are to be carried out with remote controlled, mechanized testing devices. Even here, the applicant sees the possibility of manually carrying out these tests in regions of low radiation.

In the above mentioned tests, the acoustic irradiation directions or the testing frequencies are selected in such a manner that any existing defects in their most probable orientations can be displayed.

On the accessible surfaces of the pressure vessel unit, besides the already mentioned testing procedures, the magnetic powder testing method, dye penetration-test and visual inspection can in addition be typically used as for checking the material surfaces. The visual inspection is aided by optical instruments or television cameras only if the test region is difficult to access.

The initial pressure test on the pressure vessel unit in the HTR-module- power facility takes place by using the water pressure test, which is 1.3 times the design pressure. The periodical pressure tests on the pressure vessel unit are done on the primary side by using the gas pressure test, which is 1.1 times the design pressure. Helium is used as the pressure medium. The possibility of carrying out the periodical pressure test with 1.3 times the design pressure is excluded by the applicant with reference to the conventional set of rules.

According to the conventional set of rules, an internal inspection must be done before the gas pressure test. As the internal inspection cannot be done before the gas pressure test for the HTR module, as a substitute a comprehensive nondestructive testing is specified before and after the gas pressure test /U 2.6.2–30 /.

The secondary side water pressure test is carried out with 1.3 times the design pressure.

Temperatures of the pressure vessel walls are less than 50°C for the pressure test on the primary side. For the secondary side pressure test, the temperature is above 33°C.

The practicability of periodical tests depends on the feasibility of the place of inspection for the application of tests based on the specified testing techniques, accessibility to the pressure vessel unit, its insulation and manipulating techniques. In this regard, the applicant executes the following:

- The accessibility is guaranteed by steel platforms, ladders and steps. Adequate space is available for the assembly of guide rails for manipulators. Installation of manipulators is not obstructed by space considerations.

- The test ranges are equipped with quickly removable insulation segments.

- Manipulators and manipulation recording techniques are applied which have been proved already in testing applications in light-water reactors. Required component-specific new manipulator systems are further developed on the basis of the known systems and applied. This applies, for example, to the testing of the top part of the reactor pressure vessel. Here the manipulator technique existing for the external testing of boiling water reactors is to be modified and applied.

The mentioned test interval is eight years for the execution of periodic tests for pressure testing and four years for the non-destructive type of tests.

For evaluating the possibilities mentioned by the applicant in the 2.6.2–28, U 2.6.2–31/ documents with respect to the testability of the pressure vessel unit in the case of periodic tests, we have established the basic requirements on the basis of evaluation criteria mentioned below:

- Safety Criteria for Nuclear Power Stations /L 6/,
  Criteria 1.1, 2.2, 4.1,

- KTA Rule /L 48/,
  Parts 2 through 4,

- Technical Rules for Pressure Vessels /L 30/,
- Steam Boiler Ordinance /L 4/,
- Pressure Vessel Ordinance /L 5/.

On the basis of the spatial conditions in the primary cell and design factors, we in principle approve the statements of the applicant that an adequate accessibility to test ranges is present.

The design characteristics of the pressure vessel unit are given in the /U 2.6.2–28/ documents. We have already expressed our stand in Section 2.6.2.1 with regard to the design implementation and we have also basically considered therein even the requirements of periodic tests. Details pertaining to the design and execution (*e.g.*, surface quality), which must take into consideration the requirements of periodic tests, can be established within the framework of constructional planning. The design execution and the materials required for the manufacture of pressure vessels and welded joints can basically be tested with the specified testing methods and testing devices. Proof for the viability of periodic tests with the required testing techniques and testing devices must be brought out in detail before the initial operation of the nuclear reactor with the help of suitable test and adjustment blocks, which relate to the component dimensions. We will treat separately the testing of the creep stressed regions of the steam generator pressure vessel.

For the non-destructive testing /U 2.6.2–1/, the applicant mentions the range, scope and intervals of testing. Basically all the welded seams and regions of increased stress conditions are tested. In the document, testing of bolts and threaded blind holes of the blower pressure vessel has not been taken up. Furthermore, testing of nuts of screw joints are missing. These tests are necessary under certain circumstances and they can be carried out with the eddy current method.

Up until now, no periodic tests are presented for the welded seams of the inspection doors and Helium outlet and Helium inlet connecting pieces. This is valid also for the manhole of the reactor pressure vessel as opposed to the manhole of the steam generator pressure vessel. These tests are, however, necessary in some cases and they can be carried out with the ultra-sound method.

In the test plan of periodic tests, the testing of welded seams and possibly other test-worthy regions of the blocks of fittings, such as for example those with increased stress conditions, must be taken up.

The applicant has named regions of increased stress condition, which are to be tested. From our experience in installations with light-water reactors, we can in principle approve this selection without considering the analytical results and experimental tests. After completion and evaluation of the above-mentioned results, if further regions are found, then they must also be taken up in the test plan. If the already available testing devices do not facilitate the testing of these additional regions, then special manipulators are to be used. In our opinion, this is possible with the present state-of-the-art of technology.

Within the manufacturing framework, if areas of deficiencies are noticed and left as such, or there are areas for improvements, then these must be taken up within the scope of testing.

In the Safety Report, the applicant does not rule out the possibility of execution of manual tests. However, we are of the opinion that, as far as possible, the tests described in the /U 2.6.2–28 / document can be carried out with remotely operated mechanized testing devices. Preference is to be given to mechanized testing even from the standpoint of the reproducibility of test results. The reproduction of instrument setting parameters and instrument characteristics are advantageous in the mechanized testing, and the connected computer systems facilitate a complete recording of original data and time-independent signal processing and execution of analyses. Furthermore, our experience shows that in a manual test, which is supposed to have approximately the same predictive power as the mechanized testing, considerably more time is spent.

We have no objections with regard to the execution of the secondary-side periodic water pressure test with 1.3 times the design pressure. Basically, this is valid even for the first pressure test of DBE in the installation with 1.3 times the design pressure (hydraulic test).According to the engineering documents pertaining to the periodic tests /U 2.6.2–31/, the blocks of fittings for the fuel element, rupture exclusion, for the small spherical shut-down system and fuel element feeding are mounted on the connecting pipe only after component testing. However, in our view, the fitting blocks must be mounted before the pressure test so that even the parts stressed during operation and welded seams are stressed within the framework of component pressure testing.

One of the periodic pressure tests as a gas pressure test with 1.1 times the design pressure is approved by us because even with this test pressure, from the standpoint of rupture mechanics, an adequate stress of the possibly present defective spots is achieved due to the reduced strength level at the testing temperature of below 50°C that is reduced compared to the operating temperature.

We have no objections to the testing interval of eight years for the execution of the periodic pressure test and four years for the periodic non-destructive test.

The manholes present in the pressure vessel unit are basically meant to be used for any maintenance jobs. An internal inspection of the pressure vessel is not provided even in parts of the areas. As a substitute the applicant has provided extensive U.S. tests before and after the pressure test. Among other things, this is a measure, which justifies from the safety point of view to drop the demand for an internal inspection because it promises very little predictive results due to the operative boundary conditions.

For the materials in the high temperature range special testing methods are needed to be able to assess the creep behavior and the remaining life duration of the materials within the framework of periodic tests. Therefore, the applicant must develop a test program for the periodic tests of materials and material combinations for those components, which are subjected to a creep stress at high temperatures (≥350°C).

In this context, the methods mentioned in the literature /L 106/ for the determination of remaining working life can be used. Magnetic and ultrasound methods together with the results of metallography appear as apt to us. We expect that in the framework of further development, test results offering better predictions in respect of determining the remaining working life are obtained. In a test plan, the results should lead to the creep stressed regions of the pressure vessel unit.

If the above mentioned instructions are considered in the execution of the pressure vessel unit, the basic demands of conventional and nuclear engineering set of rules, especially the KTA Rule 3201 can be maintained and the established BMI criteria can be fulfilled.

2.6.3 Hot Gas Pipe

The hot gas pipe is used for the carrying of the primary coolant between the reactor pressure vessel and the steam generator pressure vessel. It consists of the core vessel connection within the reactor pressure vessel, the straight hot gas pipe within the intermediate pressure vessel and the hot gas elbow within the steam generator pressure vessel. The suitability of the selected execution is proved by the statements of the applicant based on the previous results from the development project "Nuclear Process Heat". Tests with Helium temperatures of 900–950°C have been carried out for this. New components had to be developed because of the high Helium temperatures. These components were tested in a component test cycle. The applicant placed before us a document /U 2.6.3–1/ on the operating experiences obtained, additionally in which he examines the basic assumptions for the construction of the hot gas pipe.

The requirements for the hot gas pipe and the construction are explained in the Safety Report and in document /U 2.6.3–2/. According to that, no safety regulations significance must be attached to the hot gas pipe. Maintenance and service measures are not required according to the opinion of the applicant based on the construction and the arrangement of the hot gas pipe. Recurring checks are not planned. The interchangeability of the hot gas pipe is, however, in principle possible.

According to the system description, the hot gas pipe is neither required for the afterheat removal nor are releases of radioactive materials to be taken care of in case of an assumed break-down.

The cold gas, which flows in the intermediate pressure vessel around the hot gas pipe, has a higher pressure than the hot gas in the hot gas pipe. In case of a leak in the hot gas pipe, the cold gas enters the hot gas pipe as a consequence, and hence no hot gas can reach the pressure bearing side of the intermediate pressure vessel.

The applicant has classified the hot gas pipe as a component of the intermediate pressure vessel. The requirements for such components with individual KKS-designation are specified according to the Safety Report and document /23–1/ corresponding to their functional significance in the construction plan.

The hot gas pipe has an operative function. Leakages in the hot gas pipe would, however, require comprehensive maintenance measures in areas with high dosage rate. On our part there do not exist any doubts regarding the line of action of the applicant to determine the quality control measures within the scope of the construction plan only. Basically, we are of the opinion that, on the basis of the previous test results available, the production and operation of the hot gas pipe with their specific functional requirements are possible. However, this is possible only if adequate quality control measures are put in place. We do not consider the recurring checks for the verification of the integrity of the hot gas pipe as necessary during the operating time of the HTR module.

We expect that the applicant will determine the procedure for possible replacement of the hot gas pipe in the construction plan.

2.6.4 Steam Generator Installations

Objective

The 700°C hot Helium is supplied to the steam generator at the upper half of the heating tube bundle with an entry pressure of 60 bar through a reversing device. From the reversing device, the Helium flows down through the helically coiled tube bundle. The flow is restricted to the inside by the central pipe, to the outside by the flow casing. The Helium delivers heat by the through-flow to the water on the secondary side flowing from bottom to top or to the steam. The fresh steam produced has a temperature of 530°C and a pressure of 190 bar. After leaving the heating pipe bundle, the flow is reversed by 180°C and the Helium, cooled to 245°C, now flows up in the annular gap between the steam generator jacket and the pressure vessel wall. The heating bundle, including its support structure, the flow casings and the heat insulation, is mounted as hanging in the enlarged area of the steam generator pressure vessel, by which an unobstructed heat expansion of the individual parts is possible.

Design

The stability design of the heating pipes and plates is calculated for a lifespan of 32 full load years.

The maximum pressure difference between the operational media present in the operation in accordance with the requirements is taken as the basis for design pressure. The pressure difference occurring for a short period due to the pressure release imbalance in the primary side is also considered.

The maximum average pipe wall temperature is selected as the design temperature. For determining the design temperature, an extra 50 K for the Helium temperature and 40 K for steam temperature are provided for improper loading and for measurement and guideline inaccuracies.

In consideration of the wall thickness tolerance, an extra 10% is provided on the required wall thickness. An extra tolerance for corrosion is not considered.

The shocks induced due to earthquakes, aircraft crash and gas vapor cloud explosions are considered as dynamic loads for the design of the pipe bundle. Further, the pipe bundle is designed against pressure waves and pressure differences, which result from the assumed breakdown of the secondary pipelines. The load factors to be considered for the strength-wise design of the components belonging to the primary gas inclusion, their frequencies, their classification according to KTA Rule 3201.2 /L 48/ as well as the allocation for the demand levels are summarized in the load collection /U 2.6.4–2/.

Extensive analytical and experimental investigations /U 2.6.4–1, U 2.6.3–1/ have been carried out to safeguard the design for vibration. The measurements of the test components for the HTR module steam generator corresponding to the previous results available do not indicate vibrations that are dangerous to the structure.

Construction

The helically coiled tubes of the hot pipe bundles, which are arranged at constant radial distance to the central pipe on the cylindrical surface and whose length is almost the same, are held in carrier plates which are arranged radial to the central pipe and welded to it. Beyond that, "floating" plates are provided for increasing the pipe natural frequency, which join several tubes with one another, but are not welded to the central pipe.

In order to protect the hot tubes against wear in the area of the passage through the carrier plates and the arched floor of the steam generator head space, the wedge/sleeve design, employed successfully for the steam generation for Fort St. Vrain and the THTR as well as for the PNP intermediate heat exchanger, is used.

Since the pipe wall thickness of the hot pipe substantially influences the area of the hot surface and thereby also the pressure loss, tubes with greater wall thickness are employed for the same outer diameter above an average pipe wall temperature of 400°C.

For ensuring the flow-through stability of the steam generator for the entire load range and for adapting the water-side through-flow distribution for the different flow resistances of the individual tubes, interchangeable, stabilizing restrictors are screwed in the holes of the feedwater pipe plate. The results of the previously carried out, but not yet completed, investigations for the stability behavior do not indicate that any instabilities are to be expected in the tubes /U 2.6.3–1, L 113/. Especially still incomplete are the investigations on the influence of the cross-mixing at the inner and outer pipe cylinder and the validation by trials of the analytical investigations for the dynamic stability.

For diverting the hot Helium in the horizontal hot gas pipe by 90° to the bottom in the steam generator, a pipe bend with pivoting plates is inserted for equalizing the gas flow and minimizing the pressure loss. At the outlet of the hot gas bend, the hot gas is radially diverted outwards over a grid, which is in the form of blinds, with little pressure loss and distributed equally at the bundle cross section. The investigations are not yet completed, but it is clear that a relatively well-balanced flow profile can be achieved in the pipe bundle through constructive measures. The in-flow area (including blinds of the first pipe side in the bundle) should be investigated in a further trial on a model, scale 1:3.

Materials

The material Incoloy 800 (X 10 NiCrAlTi 32 20) is specified to be used as material for the heat exchanger tubes as well as for the feedwater and fresh steam pipe plates because of favorable mechanical properties in the high temperature range and adequate corrosion resistance /U 2.6.4–2/. The same material was used also for the heat exchanger tubes for THTR-300. Because of the higher strength values, the material is used in the soft-annealed condition at the low temperature range and solution heat-treated condition at the higher temperature range.

For the material in the solution heat-treated condition, the creep strength values are taken from 550°C based on the 100,000 hour values contained in the VdTüV Material Data Sheet 412 /L60/ and the ISO-extrapolations /L 70/ up to 250,000 hours as well as based on the additional extrapolations of the applicant up to 280,000 hours=32 full load years, are assumed. A new evaluation of all investigation data presently available /L114/, which is included in a DIN-draft /L 33/, shows that the values given in the VdTüV Material Data Sheet 412 must be corrected as follows when applied to the creep strength values. The discussion on the determination of the final valid values is not yet currently completed in the corresponding committees. The extrapolated creep strength values for the time range 100,000 hours to 280,000 hours are to be further validated based on the results of the current investigation program.

Leakage Assumptions for Steam Generation Tubes

In incident analyses, the applicant has postulated the complete breakage (2F) of a steam generator pipe as being the greatest leakage of a HTR-module-steam generator. For the postulations "Breakage of the Fresh Steam Pipe or Breakage of the Feedwater Pipe", the applicant rules out consequent damages to the steam generator tubes, since they are appropriately designed against the loads (pressure differences, pressure waves) resulting from these assumed incidents.

In case of the incident analysis for the fresh steam pipe breakage, the applicant does not assume, deviating from the DWR-guidelines of the RSK /L 10/, any 2F-breakage in the steam generation tubes as accidental, not as a result of additional error arising from the fresh steam pipe breakage /U 2.6.4–3/. This is based on the fact that for the HTR-Module substantial detrimental actions can be ruled out for the steam generator pipes in the pressure water reactors, the steam generator tubes are designed against the loads in case of a fresh steam pipe breakage and operational leakages at the hot tubes leads to a shut down of the plant via the primary side humidity measurement /U 2.6.4–3/.

The reasoning is giving detail in an additional document /U 2.6.4–4/. Accordingly, many detrimental actions occurring in steam generators for pressure water reactors, like surface corrosion (wastage), pipe constriction (denting), pitting and crevice corrosion, are not to be assumed through the selected construction and the operative boundary conditions for the HTR module. According to the opinion of the applicant, damages through erosion can also be ruled out because of using Incoloy 800 as the pipe material. Material and production errors are avoided through specified quality assurance measures. The remaining possible detrimental actions owing to vibrations, such as fretting and fatigue, were taken into account for the constructional arrangement and design.

The applicant considers the assumptions made based upon the incidence considerations as confirmed by the prior positive operative experiences with gas heated steam generators as well as by the experiences gained from experiments for the PNP-project.

The safety criteria for nuclear power stations and the RSK-guidelines are taken as basis for criterion of valuation and assessment for the technical safety requirements. The steam generator pipe bundle including feedwater and fresh steam pipe plate belong to the pressure carrying enclosure of the primary gas and are assigned the highest quality requirements corresponding to the technical safety regulations of the quality class. The components must fulfill the requirements of the KTA Regulation 3201 corresponding to the classification concerning safety-technological regulations.

The pressure differences and maximum average pipe wall temperatures taken as basis for the strength oriented design by the applicant consider the operational loads and loads due to incidents as sufficient enough for the design assessment.

Moreover, the maximum average pipe wall temperature can be agreed upon to be the design temperature if the design based on the strength is demonstrated to be adequate by detailed stress and fatigue analysis. Possible exceeding of the design temperature for individual parts, like for example, at the fresh steam pipe plate in case of breakdown in the feedwater supply (see Section 5.3.3), are to be included in the load bundle and to be considered for the stress and fatigue analysis to be carried out.

There are no objections against the use of the material Incoloy (× 10 NiCrAlTi 32 20) for the specified operative temperatures if the creep strength values are corrected downwards based on a new evaluation of all available investigation data /L 93/ compared to the VdTüV Material Data Sheet 412 /L 69/ and are taken as basis for the design based on strength.

Further, the creep strength values, extrapolated for the time range 100,000 hours to 280,000 hours, are further validated by the results of the current investigation results.

We agree with the assumption of a 2F-breakage of a steam generator pipe as the maximum leakage area for incident analyses for the design consideration, the material used, quality assurance measures specified for the manufacture, as well as the previous operational experience and investigation results. We are of the opinion that consequential damages to the tubes are not to be assumed for assumed breakages to the feedwater and fresh steam pipe for the specified design of the steam generator pipe compared to the loads resulting from the same type of breakages. Beyond that, we agree that the 2F-steam generator pipe breakage is not combined with the fresh steam pipe breakage, since there are a limited number of detrimental actions for the HTR-steam generator compared to that for steam generators in pressure water reactors, for compliance with the relevant water

chemistry and as the specified humidity measurement already detects very small leakages.

Contrary to the opinion of the applicant, we feel that it is necessary to make repeated non-destructive checks as a safety measure to be able to comply with a safety statement on the quality condition of the steam generator pipe during operation. The possibility for the performance of repeated checks on the heating pipes according to the turbulent flow checking procedures can be ensured through a respective design layout. Respective investigations with the turbulent flow checking procedures were already carried out for experimental components with positive results. The amount of inspection can be determined based on the construction plan.

We agree with the concept regarding the design, construction and material selection of the steam generator components under consideration based on previous experiences with components of similar construction type and operative conditions.

2.6.5 Primary Circulation Blower

The primary circulation blower is arranged in the upper area of the steam generator pressure vessel and is built with a vertical shaft. The single-stage rotor is overhung. The shaft is supported in oil bath bearings. The penetration of oil in the primary cooling medium circulation should be prevented through /U 2.6.5–1/ constructional measures. The entry of oil vapors in the primary cooling medium circulation cannot be prevented, however, it is very low based on the conservative estimates according to the applicant's statement.

The blower increases the pressure of the primary cooling medium, Helium, in the primary cooling circulation by maximally 1.5 bar for an outlet pressure of 61 bar /U 2.6.5–2/. It sucks the cold gas via the cold gas pipe connection into the steam generator pressure vessel and conveys it via the intermediate pressure vessel to the reactor core. The cold gas pipe connection to the inlet socket of the blower takes place through a positive sliding connection. Only the suction housing, the blower flaps, the rotor and the diffuser are in the direct flow of the cold gas.

The diffuser consists of two parts, the inner and outer diffuser. The inner diffuser is disassembled simultaneously on removal of the blower the outer is connected to the steam generator pressure vessel and is retained in the vessel.

The blower is built as an insertion unit, which can be removed as required from the blower pressure vessel, *e.g.*, for repair. Moreover, the blower pressure vessel covers are removed and the stud bolts, which join the intermediate flange of the blower with the flange of the steam generator pressure vessel lying inside, are loosened. The blower is designed to be an insertion unit, which can be disassembled if required from the blower pressure vessel, *e.g.*, for repair. Moreover, the blower pressure vessel covers are removed and the stud bolts, which join the blower intermediate flange with the steam generator-pressure vessel flange internally, are loosened.

The intermediate flange separates areas, having temperature difference of 190°C under normal operation, from each other. Above that, the motor room is located where the temperature is restricted to 60°C through water-cooling. Below that the cold gas is supplied at 250 °C. Under normal operation, no pressure difference exists between the two spaces, since the labyrinth seal of the shaft makes pressure compensation possible.

Only the blower switching-off has a safety regulation corresponding to the Safety Report. Its functioning is triggered by the safety assembly. No inspections and repeated checks are planned for the blower besides the operative monitoring. Continuous vibration and bearing temperature measurements are taken for the regular monitoring /U 2.6.5-3/.

The applicant proposes a quality assurance plan for the blower manufacture, which provides for the material selection according to reference sheet, the U.S. inspection of the forged shaft, the volumetric inspection of all stressed welding joints and a functional test at the manufacturer's premises. In this context he refers to the positive experience gathered during the operation of similar blowers in British power plants. The transferability of the positive experiences should be ensured by adaptation of the design principle of the blowers of the HTR-Module-power plant and by selection of the same manufacturer.

Although previously no damages have been caused due to the breakage of parts of such blowers, the applicant does examine possible consequential damages /U 16/. Moreover, he determines through energetic considerations that the power that may occur can be dissipated safely.

The applicant has considered the primary circulation blowers as built-in part according to the document /U 17/. Consequently, requirements for quality assurance measures are determined based on the functional significance within the framework of the construction plan.

We have taken the documents /U 2.6.5–1 to U 2.6.5–4/ as the basis for our assessment of the primary circulation blower. We cannot evaluate the constructional design of the blower in detail on the basis of the present documents. This is also not necessary in the context of the assessment of the safety regulations as the applicant has proved the suitability of the primary circulation blower essentially based on the fact that such blowers have proved themselves in existing plants. However, we consider an understandable evaluation of the operative experiences obtained in the sense of a qualified proof of operational excellence as necessary for the construction plan. Additionally, the requirements with regard to a function-related, stress-related, and inspection-related and maintenance friendly execution must be written down in a relating specification.

The computations submitted for the component failure are adapted as basis for the design assessment, in order to prove that an unacceptable damage to the pressure carrying enclosure is not to be expected. Analytical proofs are to be worked out in the context of the construction plan for representative failures, *e.g.*, even with regard to a shaft breakage. We consider as suitable the planned continuous monitoring of the primary circulation blower during the operation by means of vibration and bearing temperature measurements so as to recognize possible damages ahead of time.

Periodical tests of the blower are not provided for by the applicant. We do not see any reason for a requirement of periodical tests in the context of the safety-technological design assessment.

2.6.6 Support of the Pressure Vessel Unit

The support of the pressure vessel unit inclusive of the design loads to be dispersed is described comprehensively in the Safety Report /U 1/. Detailed information on design, material and on the behavior of the support for operational loads and for loads from anomalous incidents are available completely in the /U 2.6.6–1/ and /U 2.6.2–9/ technical documents.

Objective, Design and Layout

The support design of the applicant provides for the fact that on the one hand the operational loads and loads caused by anomalous incidents can be dispersed on the concrete structure and on the other hand an unobstructed heat expansion of the pressure vessel unit is possible. The support design was selected in a way that while the pressure vessel unit is fixed against external forces in the structure, the radial and axial displacement due to the temperature expansion processes is not obstructed. In order to achieve this objective, the pressure vessel unit is supported at three levels whereby the RDB is supported in a way that the container axis is fixed axially and radially in the structure. For that purpose, the RDB is supported in the axial and radial direction at the middle support level by three supporting brackets welded at the periphery. The supporting brackets are supported on slide bearings in a way that they permit the radial heat expansion of the reactor pressure vessel but prevent the movement in the peripheral direction. The slide bearings are provided with a graphite-based lubricant, whose durability against temperature and radiation effects is well-established, for reducing the friction.

The slide bearings are further placed in supporting core pads, which are anchored on the walls of the reactor cavern. In order to limit the temperature of the concrete in the area of the supporting reactor pad to admissible values for concrete, the concrete is provided here with water-cooling. According to the applicant's statement, this design has already been implemented in the nuclear power plant at Attucha. In order to prevent a tilting of the reactor pressure vessel around its middle level position, four guides equidistantly distributed from the periphery are fixed at the upper support level. These guides are arranged horizontally so that they deviate from the tangential direction by about 15°.

The steam generator pressure vessel is supported at the middle level by two supporting brackets, so that the vertical loads are borne by the concrete through slide bearings on supporting core pads. The slide bearings are designed in a way that the temperature displacements of the steam generator vessel in the lengthwise direction of the intermediate pressure vessel and radial heat expansion of the steam generator vessel are possible without expansion problems. One more additional guide is provided at the mid level, which supports the steam generator tangentially through a support in the concrete, for taking up the horizontal forces, which can act on the steam generator perpendicular to the axis of the intermediate pressure vessel. Additionally, two tangentially arranged guides are included in the plan at the lower support level for this purpose, which lead forces perpendicular to the axis of the intermediate pressure vessel to corresponding support locations in the concrete.

In order to avoid bending stresses of the intermediate pressure vessel caused by temperature related expansion of the pressure vessel unit, a maximum unobstructed displacement possibility for the steam generator must also be provided in the vicinity of the lower support level. On the other hand, dynamic forces caused by induced shocks, which would lead to very high stresses in the intermediate pressure vessel, must be avoided. This is achieved through two shock brakes, which produce a nearly rigid connection between the steam generator and the concrete structure for dynamic mechanical actions, but otherwise make an unobstructed displacement possible.

In our opinion, the planned support design of the pressure vessel unit is suitable for dispersing all designed loads, in the vertical and horizontal direction, for tare weight, operation, and anomalous incidents. For that, all elements of the support constructions, like *e.g.*, supporting brackets, slide bearings, supporting core pads, guides and shock brakes, must be dimensioned in the construction plan depending on the required loads. We do not envisage any design related problems here, since the loads for the dimensioning of the support elements can be accurately determined by mechanical analysis. A dimensioning of the elements of the support design can be carried out with the designed loads without difficulty depending on the technology status, since the effectiveness and the functionalities of the elements of the supports are well-known, as far as these have to disperse these forces.

In principle, the support design considered also ensures that the pressure vessel unit permits unconstrained heat and temperature expansions. The slide bearings considered for that in the reactor pressure vessel and the steam generator pressure vessel represent proved designs, whereby through careful design of the sliding surfaces and corresponding selection of the lubricant unwanted constraining forces are avoided. Further, the selected arrangement of guides at the different support levels enables a sufficient displacement possibility of the vessel in the areas in which this is necessary to avoid undesirable forces caused by constraining forces.

It has to be demonstrated how the radial and axial temperature expansions of the reactor pressure vessel at the upper support level are compensated through proper selection of the fitting position of the guides so that a constraint-free operation is achieved. It must also be investigated as to how the constraining forces develop in the cases of anomalous incidents. For such investigations, the wall temperatures of the reactor pressure vessel and the guides for different plant conditions are to be considered. Moreover, advantage could be taken of the various functions of the support joints. Analogously, such verifications could also be mad for the guides of the steam generator pressure vessel.

An assessment performed by us demonstrates that the selected support design guarantees sufficiently constraint-free support of the pressure vessel unit whereby the continuous safety against external forces is ensured.

Design

The design includes following loads cases:

- Self-weight of vessels, components and support structures,
- Temperatures (ambient temperature, temperatures at the contact locations between support and components, temperature gradients and transient temperature changes during the operation and in case of disturbances),
- EVA load cases,
- EVI load cases.

Besides, the behavior of the pressure vessel unit is also investigated by the applicant for the following combination of events:

- Failure of the shock brakes at the lower DE-support level in case of earthquakes,
- Blocking of a shock brake at the lower DE-support level during operation from effects of obstructed thermal expansion for the pressure vessel unit.

The analytical proofs for the supports extend to the dimensioning of the claws and lugs welded to the vessel for preliminary design loads /U 2.6.6–3, U 2.6.6–4/. No calculations were submitted for the designs of attachments like journal bearings, shock brakes, guides or the structural joints. We have no objection in principle against this procedure, since the components can be realized for the design loads indicated.

The design loads in the work reports /U 2.6.6–3, U 2.6.6–4/ are confirmed through investigations of the applicant, whose results are summarized in technical documents /U 2.6.2–23, U 2.6.2–9/. The investigation results are based on finite-element calculations available to us in excerpts. The mathematical model, consisting of beams, springs and masses, is described in detail in the /U 2.6.9–2/ work-report. It considers the structure inclusive of the separately attached pressure vessel unit and also the foundation, which affects the vibration behavior of the entire structure significantly. The latter is taken into account through six springs and damper, which simulate typically and not absolutely location-related ground values (average ground modulus of transverse rigidity between 160 and 400 MN/m^2).

As standard load cases for the support, the EVA load cases have proved themselves, based on the investigations, as safe loads to account for earthquakes and aircraft crashes. On the other hand, the EVI load cases, causing interruption of steam generation or water supply pipeline, are of secondary importance.

The stresses due to earthquake were determined with the echo-spectrum method, the effects from aircraft crash with the Time History-Method. The loads on the support due to earthquake lie below the design loads, those from aircraft crash clearly even below that.

The load cases were subject to a special consideration with the assumption of a failure of /U 2.6.2-23/ shock brake. The main purpose of these investigations is to determine the maximum possible stresses in the intermediate pressure vessel from the view point of rupture exclusion.

The applicant has carried out several calculations for this, in which especially the spring stiffness of the support attachment of the VDB at the RDB and at the DE was varied, since these parameters influence the stresses in the VDB considerably. In the case " Blocking of the Shock Brakes and Obstruction of Thermal Displacements", the applicant has reported the bearing pressures at the claws, resulting from different variation calculations. Thereafter, a re-arrangement of the load takes place in such a way that a load alleviation for the DE-claws and an overloading of the two RDB-claws takes place by approximately 50% respectively.

A comparative calculation by us with a FE-model formed of shell elements has shown that the applicant's indications for the load increase at the claws are conservative and the guide loads are clearly less than the design loads.

In the case "Failure of the Shock Brakes in Case of Earthquake", the applicant has not reported any bearing pressures. According to our own calculations, the loads are clearly higher than in the case mentioned before. Besides furnishing of an analytical proof, also a limitation of the loads through constructional measures, *e.g.*, stops, come into question (see also Section 2.6.2.3). To summarize, we arrive at the conclusion that the continuous safety of the pressure vessel unit can be guaranteed for the investigated load types through the support concept presented.

Materials

The carrying consoles, guides and brackets of the supports for the pressure vessel unit are manufactured from the material 20 MnMoNi 55. The parts are manufactured from forgings.

The temperature changes that are encountered require an unconstrained expansion of the pressure vessel unit. Because of this, journal bearings are provided at the middle support level, which make possible an unobstructed movement of the claws on the carrying consoles. The journal bearings consist of self-lubricated, heat resistant and radiation-resistant graphite based sliding metal from the company Litton/Merriman. The joining elements of the brackets with the concrete structure, like anchors and nuts, are made from the material 26 NiCrMo 14 6 (for anchors) and 34 CrNiMo 6 (for nuts).

For the assessment of the feasibility and qualification of the material for the supports, the evaluation criteria mentioned in Section 2.6.2.4 are to be referred to.

The proof of feasibility was brought for the material 20 MnMoNi 55. The results of the material tests by the experts are considered in the VdTüV Materials Data Sheet /L 163/. The same applies to the materials 26 NiCrMo 14 6 for anchors and 34 CrNiMo 6 for nuts. There are no objections against the usage of the materials.

For ensuring the unobstructed function of the journal bearing of the support design, the applicant uses a lubricant, which is already known to us from nuclear plants with pressure-water reactors. We have no objections against the use of this sliding metal for the planned application area.

The applicant gives no indication regarding the material that he wants to use for the manufacture of the shock brakes. As usual, feasibility tests for the application case are carried out. The feasibility and qualification of the used material is proved in the framework of this feasibility test. This must also take place for the shock brakes provided for the HTR-module. One such feasibility demonstration can be carried out for these shock brakes.

Periodic Tests

Periodic tests are planned only for the shock brakes of the support structure according to the /U 2.6.6–1/ document. The testing interval should be selected in conformance with the revision cycles of the HTR-module.

We have taken the safety regulations of the support structure as basis for the assessment of the requisite periodic tests. We have come thereby to the conclusion that the periodic visual tests connected with the surface tests and dimensional tests are necessary.

The aim of the visual inspection is to determine the deformations, damages, corrosion and the free movement of the supports and shock brakes. With dimensional inspection, the set cold and hot play can be verified which will avoid a obstructing of the movement of the pressure vessel unit during the operation. Surface tests are carried out with the dye penetration and Magna-flux processes, if indications on surface damages are available. Therefore, visual tests, dimensional tests and surface tests are to be taken up for the supports of the pressure vessel unit in the inspection plan for the periodic tests of the pressure vessel unit.

2.6.7 Technical Process Design

The technical process design data of the nuclear steam generator system summarized below are contained in the Safety Report, in the summary drawings of the RDB and the /U 2.6.2–1/ steam generator as well as in technical documents /U 2.6.4–1, U 2.6.3–2/.

Pressure carrying wall of the pressure vessel unit:

Design pressure p_e=70 bar
Nominal operating pressure p_e=60 bar
Design temperature t=350°C
Nominal operating temperature t=approximately 250°C

Steam generator secondary side:

- Water supply fittings

  Design pressure p_e=240 bar
  Nominal operating pressure p_e=210 bar
  Design temperature t=350°C
  Nominal operating temperature t=170°C

- Live steam fittings

  Design pressure p_e=208 bar
  Nominal operating pressure p_e=190 bar
  Design temperature t=540°C
  Nominal operating temperature t=530°C

The nominal blower flow rate amounts to 85.5 kg/second corresponding to 15.7 m^3/second at nominal operating pressure.

The design concept of the pressure vessel unit or the primary circuit basically ensures that the pressure carrying enclosure does not come in contact with hot gas in order to be able to use approved materials from the light-water reactor technology.

Related requirements for the operating values to be followed arise from that for the technical process design, essentially the maximum permitted cold gas temperature of 350°C. For the selected normal operating pressure of p_e=60 bar, a hot gas temperature of 700°C and a cold gas temperature of 250°C results for the nominal load on the basis of the heat transfer properties and the corresponding through-put of the primary gas Helium.

On the secondary side, superheated fresh steam is produced with a temperature of 530°C for p_e=190 bar. The steam generator works accordingly with quite a high gradient, which leads to the steam temperature reacting very sensitively to changes of the heat flow density in the heating tubes of the steam generator. In order to avoid too high steam temperatures, for *e.g.*, through incorrect high speed of the blower, a fresh water temperature limit should be planned after clarifying with the applicant, which are in accordance with the /U 2.6.7–1/ reactor power regulation.

The selected gradation of the design values to the operation values permits sufficient free space for regulation bands and transient operation in accordance with the regulations, without the design values being reached. The cold gas flow prevents the hot gas from contacting the pressure carrying walls and hence keeps the design temperature in the operation in accordance with the regulations. The observance of the design values in case of anomalous incidents is verified in Chapter 5 of this assessment.

The design of the primary circuit pipelines connecting to the pressure vessel unit conform to that of the pressure vessel unit itself. The pipelines are connected solely in the cold gas region so that their temperature-wise design is safeguarded through it.

The double-strand pressure release system is designed in a away that the first safety valve acts at p_e=69 bar with a gas discharge capacity of 0.15 kg/second and the second safety valve acts at p_e=72.5 bar with a gas discharge capacity of 10 kg/second.

The pick-up pressure is rated such that a sufficient pressure is ensured for the pressure carrying enclosure according to the requirements of the AD-code of A2 /L 71/ practice, even after considering the pressure loss in the supply conduit. The determining factor for design of the nominal discharge capacity of 0.15 kg/second for the 1st safety valve is the anomalous incident "Steam Generator Hot Tube Leak" with the presumed lack of availability of the water separator failure and the pressure set of rules as well as keeping the blower flap open. The anomalous incident process is evaluated also with regard to the pressure release in the Section 5.4.3 of this expert's evaluation report. Consequently, the 1st primary safety valve covers the anomalous incidents that are the determining factor for the pressure safeguard on the primary side.

In order to be able to provide cover for occurrences exceeding the designed pressure safeguard, a maximum discharge area of DN 65 was selected for the 2nd primary safety valve whose discharge quantity amounts to approximately 10 kg/second. The selected nominal width corresponds to the largest connecting conduit at the pressure vessel unit so that no larger outflow area can be formed in case of a malfunction as considered during the analysis of the anomalous incident about coolant loss.

For the selection of the safety valves to be installed, it must be ensured in the design layout that type fittings with inspected components are used in order to be able to meet the requirements according to AD-code of practice A2 regarding a proven high reliability of the valve. For the primary circuit safeguard, type of fittings would be installed accordingly, whose technical data (*e.g.*, discharge, discharge coefficient, seating cross-section) are matched to the Helium medium to be discharged.

In the case of the specification "Heating Tube Leak", the primary gas Helium is contaminated by brought-in material, so that for covering the anomalous incidents determining the design for the 1st safety valve (hot pipe rupture with after-effects), a serviceability proof is necessary, which provides for the discharge of a mixture of Helium, water gas and steam for a nominal design of 0.15 kg/second. The original geometry of the supply legs must be considered for this serviceability proof.

2.6.8 Leak-tightness Precautions and Leakage Monitoring

The primary gas inclusion poses high requirements on the system leak-tightness. Therefore, removable flange connections are provided in the pressure vessel unit with welded lip seals or metal ring seals.

A leakage monitoring serves for the monitoring of the technical sealing of the system or the pressure vessel, which detects the exhaust air from the primary cell to Helium leakage. Besides this operational objective, the leakage monitoring has still another technical safety importance. The application of the leak-before-rupture criterion demands safe leakage detection. Moreover, the measuring accuracy must be high enough that even such small leak rates, like those not to be ignored beyond the operation time due to assumption of wall penetration fissures in the pressure vessel unit based on the mechanical rupture considerations, can be definitely detected. The proposed design of the leakage detection system for leak sizes of 1 mm^2 is sufficient for this /U 2.6.8–1/.

Moreover, the different thermal conductivity of Helium and air in the exhaust air of the primary cell serves as a measuring effect. The measuring device must be qualified by means of a qualification test. Further, the thermal gas movement in the primary cell must be analyzed with adequate mixing in order to prove that a representative gas mixture is present at the measuring location.

Since the rupture exclusion is brought in also for the secondary side connection areas of the steam generator pressure vessel, the same monitoring requirements for the leakage detection are to be placed for this. The areas of the water-steam-circuit are therefore monitored for humidity in the air space /U 2.6.8–2/. We consider the indicated detectable leak-area of 10 mm^2 as sufficient for safeguarding the leak-before-rupture criterion also in the secondary-side connection areas of the steam generator pressure vessel. The proof of the detectability of this leak-area must be furnished, taking into account the air frequency number of the corresponding room volumes. The qualification test for the measuring devices for the humidity measurement must be included in the constructional design layout.

2.6.9 Function During Failure-free Operation

2.6.9.1 Primary Circuit

During production operation, for dissipation of afterheat and for the start-up and shut-down of the plant, the primary cooling medium Helium is circulated with the primary circuit blower. The Helium enters through the intermediate pressure vessel into the reactor pressure vessel bottom part. From there, it is lead to the top through the cold-gas holes in the side reflector of the core components. After deviation, it flows downward through the core to the hot gas collection chamber and leaves the RDB through the hot gas pipe. This is arranged coaxially in the intermediate pressure vessel; it flows into the head of the steam generator pressure vessel and ends in the form of a 90° bend through which the hot gas is diverted to the hot pipe. The hot gas flows through the spiral tubes bundle towards the bottom casing-side, deviates from there and rises in cooled down form between steam generator casing and pressure vessel wall towards the top to the cold-gas collector. From here it is sucked up by the primary circuit blower, conveyed to the head space of the steam generator pressure vessel and from there again to the entry connecting pieces through the annular gap of the intermediate pressure vessel.

For full load operation, a cold gas temperature of 250°C is obtained at the RDB-inlet for an operating pressure of approximately 60 bar as well as a hot gas temperature of 700°C at the RDB-outlet. The Helium energy is transferred to the water-steam-cycle by the through-flows of the heating tubes bundle, whereby it is cooled down from 700°C to about 250°C. The heating of the cold-gas by 5°C up to the RDB-inlet occurs through the energy supply in the primary circuit blower.

The Helium flow in the primary circuit is arranged in a way that the pressure carrying walls of the vessel are contacted and subjected to pressure only by the cold gas (approximately 250 °C). The hot gas flow is taken over from the inner tube of the intermediate pressure vessel, which is loaded only by the pressure difference between the hot and cold gas resulting from the pressure increase through the blower. The core fittings and the pressure drop from the cold gas to the hot gas as well as high flow resistance of the side ceramic structures of the core vessel ensure that the pressure-retaining wall comes in contact only with cold gas in the inner space of the RDB. The separation of hot and cold gas occurs in the steam generator through a gas sealed pressure-column cylinder between air stream and steam generator casing. The pressure-column remains above the head space of the steam generator with the cold-gas flowing pressure side of the blower in connection.

Consequently, leakages always lead to the transfer of the cold gas to the hot gas. Hence, the Helium gas flow within the pressure vessel unit of the primary circuit permits that the pressure retaining walls struck by the Helium need not be designed for the hot gas temperature. It can be ruled out that hot gas reaches the pressure-retaining wall.

In the steam generator, the water of the secondary circuit flows upwards through the spiral heating tubes in opposite direction of the Helium. The feedwater enters below into the steam generator through feedwater connecting pieces arranged sideways and is distributed from there via the bottom tube plate on the intermediate legs to the heating tubes bundle. The feedwater entry temperature amounts to 170°C. The water flows at an entry pressure of 210 bar upwards through the spiral tubes, is evaporated and superheated and reaches via a compensating bundle to the fresh steam outlet connecting pieces at 530°C and approximately 190 bar. The spiral tubes are thereby divided into two successive ranges – evaporating and superheating – whose specific requirements are taken into consideration for the tube design.

The steam generator is a heat exchanger to be operated in forced flow with upward evaporation. For stabilizing the flow on the secondary side, stabilizing throttles can be installed before the bottom tube plate at the feedwater entry. The temperature profile within the heating tubes bundle is balanced with that. The power transfer at the secondary circuit can occur as planned in the load range of 50– 00% as well as for the start-up and shut-down (cold operation).

The head space of the steam generator pressure vessel has the primary circuit blower for the circulation of the primary gas. Its operational function, during operation according to rules – power production, start and stop operation, dissipation of afterheat– is to maintain the Helium flow rate. Moreover, the flow rate control occurs corresponding to the respective plant condition through the rpm set of rules of the /U 2.6.5–2/ drive motor. The maximum pressure increase amounts to about 1.5 bar for a feed capacity of 85.5 kg/second.

The operation of the blower has no safety-technological requirements, since a dissipation of afterheat even without Helium flow rate is ensured by heat dissipation in the RDB at the pressure retaining wall and from there by heat radiation and free convection at the surface cooler in the primary cell. Since a circulation of the primary cooling medium is not necessary for controlling the anomalous incident, the drive motor of the blower is not supplied with emergency power network.

The possibility to shut-down the blower through the reactor protection system is important from safety-technological point of view. The operation of the blower must be interrupted in case of anomalous incidents, since the safety design of the plant provides basically for the uncoupling of the primary and secondary sides from each other thermodynamically. This also deposits in the feedwater and fresh steam sides on the secondary side. The safe shut-down of the blower is done electrically by two relays connected in succession and an additional thyristor control connected in series, besides signal formation based on process control (see Sectoin 2.13.2).

Therefore, an isolation valve arranged in the suction side takes care of shut-down the Helium flow in case of blower stoppage. It is controlled in parallel by the blower shut-down. The electrical control is easily built up and is protected by emergency power.

The drive of the blower flap is in the top of the steam generator vessel, like the blower motor, above the intermediate flange. This space is filled with Helium and is under system pressure, but it does not flow due to the blower operation. This space with "stagnant Helium" is cooled by an inbuilt water cooler to approximately 60°C. The cooler is connected to the nuclear intermediate cooling system.

Besides the drive, the bearing units of the blower are accommodated in this space. The two oil-quenched bearings each have their own oil sump, to which the external oil supply system is connected, which serves merely to fill and empty.

The intermediate flange in the steam generator pressure vessel is occupied only by the blower drive shaft and by the drive shaft of the blower shut off valve. Sealing is achieved by labyrinth packing. An entry of 0.14 g/day of oil vapors into the primary circuit via this path is indicated /U 2.6.5-1/. According to the Safety Report, it is assumed that a quantity of 5 g/d penetrates into the primary circuit.

After switching off the blower, it is advantageous in the normal operation as well as for anomalous conditions, but not necessary from the safety-technological point of view, to stop small natural circulation of Helium developing in the pressure vessel unit of the RDB for the steam generator. This is made possible by the provided isolation valve.

The concept of the oil-quenched bearings is adapted for the blower to limit the entry of oil into the primary circuit to a small quantity. The quantity that has entered can be retrieved again via the Helium purification facility and does not impair the operation of the primary system. The arrangement of the bearings and the drive for the blower and the lock-up valve within the steam generator pressure vessel safeguards the primary gas inclusion, since no shaft ducts are necessary in the pressure vessel wall.

The space for the blower drive and blower flaps filled with stagnant Helium must be held at a temperature below 60°C for the admissible temperatures for drives and bearings. A continuous cooling is necessary for the heat entry of the drives and of the intermediate flange struck with cold gas of 250°C in order to prevent a shut-down of the blower via a component protection. We consider the water cooling, which is provided with a connection to the nuclear intermediate cooling system as a sufficient measure.

2.6.9.2 Pressure Regulation

A continuously operated pressure regulation is not provided for the primary circulation. The Helium stock of the primary circulation reacts indeed quickly and sensitively to temperature changes with corresponding pressure gradient, but this nevertheless does not require a fast pressure regulation.

Therefore, a two-point regulation with feeding in and out of Helium serves for the pressure regulation in the primary circuit. This occurs via connections from the pure gas storage at the Helium purification facility, which remains connected with the primary circuit during the operation.

The primary circuit pressure is maintained within the specified operational limits via the feeding-in and -out provided. The maximum feed in flow and out flow of 135 kg/hour allows a pressure gradient of 3 bar/hour. This is sufficient not only for the gradients to be expected during operation, but also for gradients caused due to anomalous incidents. Therefore, in our opinion, a variable control is not necessary. However, an over-supply and exhaustion of the primary circuit must definitely be avoided. Corresponding monitoring devices are provided in the constructional plan.

2.6.9.3 Pressure Release System

The primary circuit operating under gas pressure is, besides the nuclear regulations, also subject to the conventional regulations, especially Pressure Vessel Regulation /L 5/. The pressure release system JEG is provided for conforming to the requirements of the TRB 403 /L 30/, which requires appropriate devices for making the vessel, herein the pressure vessel unit, safe under pressure.

It is planned as two-conduit safety valve system, whose discharge and pressure release legs branch out from the extraction conduit of the Helium Purification facility KBE /U 2.6.9–1/.

The blowpipe for the safety valve device branches out with a DN 25 pipe conduit at first from the common discharge pipe of the DN 125 JEG-system, with a valve opening area of DN 10 with a discharge output of 0.15 kg/second gas for a threshold pressure of p_e=69 bar. A spring loaded expansion joint-full lift safety valve is provided for safety. A valve of the same type is connected in series to this valve, which serves as a pressure-relief valve. It should undertake the shut-down function in this safety valve leg in case of a non-closure of the safety valve after an actuation. In this case, it shuts off for a reseating pressure of 8% below the operating pressure. The first pressure-relief valve leg DN 10 blows off, if required, filtered into the atmosphere via the KLA venting. The ventilation is laid out accordingly.

Since the pressurizer valve opens at 3% below the operating pressure and consequently remains open during the power operation, only the seating of the safety valve is available as seat leak face to the outside. In order to avoid possible seat leakage to the outside happening continuously, a rupture disc is installed in the relieving capacity safety valve, which frees the opening area at approximately 10 bar. The pipe length between the valve and the rupture disc is degassed via a connection to the KTU system (pressure relieving system for the Helium auxiliary circulation and the BE-handling) to prevent a pressure build-up, which could burst the rupture disc.

The 403 TRB and the applicable AD-code A2 of practice require a periodical annual testing for the safety valves. This requires the planning of a test connection, which comes before the pressurizing valve and behind an additional stop cock. A corresponding test pressure can be applied via this test connection. Subjecting the entire primary circuit to the test pressure is not possible and cannot be done.

The common discharge pipe of the DN 125 JEG system mentioned in the beginning goes to a second conduit after the branching of the 1st safety valve leg. The arrangement is basically like the 1st conduit conceived with front-position pressurizing valve, test connection with manual shut-off and rupture disc in the discharge pipe. The layout of the 2nd safety valve leg is considerably higher with a valve opening area of DN 65 with a discharge flow of 10 kg/second gas for a threshold pressure of p_e=72.5 bar. An auxiliary controlled safety valve, which actuates in the intrinsic medium is provided as the safety valve, which works according to the discharge principle. Two control units serve for the controlling, whose pressure extraction pipe branches separately from the KBE extraction conduit, which cannot be shut off and remains uninfluenced by the discharge pipe of the main valve.

Each control unit consists of a spring-loaded controlled safety valve in the intrinsic medium, which serves as impulse valve, and an auxiliary controlled shut-off valve, which is found in the control leg belonging to the respective control unit. The function of the control units is independent of each other.

The pressure dependent impulse valve is opened in both units at a pressure of 72.5 bar. The resultant mass flow controls the closure, through which the buncher space of the safety valve is relieved. After sufficient discharge, the safety valve opens and releases the full discharge area of DN 65. The opening operation of one control unit is sufficient for the reliable opening of the main fitting, hence both the control units must be re-closed for the shutting off. The actual throughput of the impulse valve of a control unit amounts to 0.15 kg/second corresponding to the throughput of the 1st safety valve.

A single pipe, auxiliary controlled, intrinsic-medium-operated valve serves as pressurizer valve, which opens at about 3% below the operating pressure and again closes at about 8% below the operating pressure. The pressurizer valve should also take over the closing operation in this pipe in case of a possible non-closure of the main valve after the actuation.

The 2nd safety valve leg discharges like the first one into the reactor room. However, the filter plant for the ventilation is not in a position to throughput 10 kg/second, so that the discharged mass flow of the 2nd conduit is discharged unfiltered via the discharge openings.

The overall concept of the primary circuit pressure safety provides for two parallel safety valve legs with different discharge capacities, in which the first, smaller leg suffices fully for the design-basis accidents. Further, both the legs have pressurizer valves connected in series and additionally a shut-off fitting each for test and repair purposes. Although the shut-off fittings are blocked from each other in such a way that always only one complete conduit can be isolated, the system taken by itself does not correspond to the applicable set of rules on account of the pressurizer valves connected in series. This does not provide, namely, any preliminary shut-off for the safety valves. The total fulfillment of the set of rules is however achieved through the second safety valve leg. The impulse valve of each control leg represents by itself a spring loaded safety valve that cannot be shut-off, which can discharge the mass flow required for the design-basis accidents. Since two such impulse valves are present, the requirements of

the formal set of rules are exceeded by this. The common supply conduit of the impulse valves is sufficiently dimensioned with DN 25 so that no functional impairment is to be expected during the operation of both the values a free cross-section of DN 10. Disregarding the functioning of the 1st and 2nd main safety valves, a single impulse valve in a control unit of the 2nd safety valve leg is sufficient to meet the requirements of the set of rules. The failure calculations made for the 1st safety valve leg show that the transient cover-up determined is recovered without a pressure over shooter for a set pressure of p_e=69 bar with a throughput of 0.15 kg/second. Consequently, an impulse valve can also control this transient for a set pressure of p_e=72.5 bar, which lies 2.5 bar above the primary circulation design pressure within the required range up to 1.1 times design pressure.

Consequently, the proposed concept of making safe the primary circuit pressure is in a position to limit, through the two safety valve legs, anomalous incidents determining the design (1st leg) and occurrences exceeding the design specifications (2nd leg) in such a way that an individual error caused by the non-closure of a safety valve is also controlled. Further, independent of that, the requirements of the set of rules is met by the impulse valve, whose construction is same as that of the 1st safety valve and with that all required qualifications are fulfilled.

2.6.9.4 Pressure Compensation System

The pressure compensation system consists of an external, shut-off pipe line DN 65, which connects the steam generator space on the pressure side of the blower with the head space of the RDB that is under stagnant Helium together with the annular gap between the core vessel and the pressure-retaining wall. The manual shut-off is operationally open and is closed only for ambient pressure in the primary circuit in the shut-down condition of the plant.

For the operation according the design, the core vessel casing is surrounded outside by Helium at a higher pressure level (approximately 1 bar) than inside, which is achieved by this arrangement. By this, a selective leakage flow is created as long as a leak is present.

In the shut-down condition of the plant, the stagnating Helium region can be separated from the remaining part of the circulation by means of the manual shut-off in the pressure compensation line. The manual shut-off provided is sufficient for this operational objective. Beyond that, the off-position of the manual shut-off must be safeguarded in the plant operation according to the rules. This is necessary, since the design of the core vessel casing does not permit a pressure difference greater than about 1 bar.

The safety regulation function of the pressure compensation system is to secure the pressure compensation between the throughput and the stagnating Helium region for pressure loss failures. In this way, pressure compensation can be ensured for leakage magnitudes up to diameters of 30 mm in the throughput and of 20 mm in the stagnant region.

Additionally, in the RDB, an internal pipe line DN 200 is provided from the core vessel to the head space of the RDB, which has in the head space two bursting control devices arranged in parallel effectively opposite to each other. This device is adapted to initiate pressure compensation procedures, which cover the maximum leak cross section of DN 65 corresponding to the design-basis accidents. After a burst control operation, the required shut-off can be taken up later via the slider in the head space of the RDB.

2.6.10 Operation in the Case of Failures

2.6.10.1 System-internal failures

In this section, those failures, which can result from the primary circuit and have effects on the entire plant, are handled.

Anomalous Incidents due to Coolant Loss

The leaks to be considered based on the rupture assumptions as a result of

- Tearing of a DN 65 pipe behind the primary circuit closure fitting,
- Tearing of a DN 65 pipe between pressure vessel unit and shut-off fitting (unstoppable leak),
- Tearing of small conduits, e.g. DN 10 slotted measuring section,

and small leaks are analyzed in Section 5.4 of this assessment with regard to their effects on the entire primary circuit and on the observance of the protection limits. The radiological effects are considered in the Section 5.8.2 of this assessment.

Anomalous incidents due to coolant loss all the way to tearing of slotted measuring sections are detected by the following two suggested criteria in the reactor safety system:

- Negative variable limit value of the primary circuit pressure more than/equal to 180 mbar/minute
- Throughput ratio of the primary side to secondary side is less than/equal to 0.75.

The protection actions – quick switching off of the reactor (RESA), switching off of the blower, shut-off of the secondary circulation and primary circuit shutting off – are initiated according to the rules for achieving this limit value. The closing of the blower flap is also connected with the switching off of the blower.

For detecting small leaks, the leak monitoring system, the balancing of the Helium feeding in and out from the pure gas storage and the space air monitoring can be used. Consequently, corresponding manual measures for the shut-off of the leak location or for the start-up of the plant can be introduced.

The disturbance analysis in Section 5.4 shows that basically all leak locations and sizes are controlled. The necessary residual heat dissipation after the disturbance occurs via the flange cooler because of the secondary circuit stoppage.

Malfunctions in the Pressure Release System

Basically the non-opening of the valves and the non-closing after response are to be considered as malfunctions in the pressure release system.

For a sufficient pressure release, the opening of the 1st safety valve is necessary in the design case to be considered. Should this valve fail, not only the 2nd safety valve but also its two control lines are available as redundant devices, since a sufficient cross section is released via each individual valve. The around 3.5 bar higher response pressure of the 2nd safety valve leg has no substantial influence on the set disturbance-process.

The non-closing of a safety valve after response is to be treated as individual error. This disturbance is limited through the pressure maintenance valves connected in series, which closes automatically at 8% below the operating pressure. The previous leading event for the response of a safety valve would be enlarged to a short time, isolatable primary leak. Should the corresponding pressure maintenance valve also not close or one of the two control lines of the 2nd safety valve not close with 0.15 kg/second throughput, a leak that cannot be isolated would arise in the primary circuit with a maximum free cross section of DN 65. The disturbance effects are sufficiently covered by the analysis for the non-isolatable leak with a maximum cross section of DN 65. Moreover, an open safety valve leg can be isolated by closing the manual shut-off fitting in the discharge pipes required for test purposes as medium term solution. This applies also for the control lines of the 2nd safety valve. By this, the free settings can be reduced and minimized.

The erroneous opening of a safety valve out of the production operation is covered by the considerations for the non-opening of a valve.

Steam Generator Heating Tube Leak

The steam generator heating pipe represents the essential barrier between the water and steam-carrying secondary side and the Helium gas atmosphere of the primary side. Leakages lead to the water or steam transfer to the primary side by the pressure gradient from the secondary side (p_e=190 bar) to the primary side (p_e=60 bar).

The double-ended tearing of the heating pipe is considered in totality for the steam generator heating tube leakages. Hereby, the overflowing water quantity is restricted by the throttle inserted in each feedwater sided feeding of a heating tube. The water break-through in the primary circuit is detected via a suggested criterion "Humidity in the Primary Circuit More Than/Equal to 800 vpm".

The RESA protection actions, blower switching off, secondary circuit closure and steam generator unloading are initiated in case of detection of a heating tube leak. The total water quantity crossing over can be so restricted by these measures that the estimate of 600 kg water is conservative.

The disturbance process is handled in Section 5.4.3. The disturbance is controlled observing all design values. Moreover, additional failures of operation systems were suppressed, whose function could minimize the disturbance consequences.

The residual heat dissipation takes place via the flange cooler. A cold-operation of the reactor via the disturbance-water separator is basically possible by its availability.

Failure of Blower

The operation of the blower, necessary for the circulation of the cooling medium, depends essentially on the current supply and the cooling of the upper head frame of the steam generator pressure vessel, in which the drive and the bearing units of the blower are arranged. Anomalous incidents in the current supply can be brought about by failure of the supply lines, error in the switchgear, or also by error in the supply technique for the blower switching.

A disruption of the cooling of the upper head space would be attributed to a malfunction or failure of the operating component cooling system. In all cases, the consequence is the failure of the blower.

A breakdown of the drive shaft, the bearings or the rotor of the blower also leads to the malfunction of the blower and with that to the disruption or interruption of the primary cooling medium throughput.

The failure of the blower leads to the interruption of the primary cooling medium throughput. The detection takes place through the suggested criteria

- Throughput ratio (primary to secondary side) smaller than/equal to 0.75, and
- Negative sliding limit value of the thermally corrected neutron flow more than/equal to approximately 20%/minute,

and leads to the initiation of the protection actions RESA, secondary circuit closure and switching off of the blower as well as for the operational closing of the blower flap. The disturbance process is covered by the process according to RESA. It is described in Section 5.3.1. All design values of the components are observed.

Incorrect Closing of Blower Flap

The wrong closing of the operational blower flap, which is arranged on the pressure side of the blower and should cut off the remaining natural circulation in the primary circuit in case of a blower standstill, is made possible by manual or control technological incorrect excitation.

The closing of the flap in the output operation brings the cooling medium circulation to a standstill like a failure of the blower. The interruption process is comparable to that of the blower failure. It is controlled by the plant, all design values are observed. The event does not result in the switch-off condition "hot, sub-critical", since the closed condition represents the predetermined condition. However, the flap must again be opened for the cold-operation of the plant.

Non-closure of Blower Flap on Demand

The blower flap is operationally closed for all shut-downs of the blower in order to completely stop the primary coolant flow. A non-shut-off of the flap leads to a small natural circulation in the primary circuit, which does not have any significant effect on course of anomalous incidents. Hence, the blower flap has no safety-technological significance.

Faulty start-up of the blower

The primary circuit blower has variable-speed in the entire output range in order to thus control the Helium throughput and the hot gas and cold gas temperature associated with that according to the specified partial load diagram.

Errors in the electronic control cause either a reduction in the speed, which is comparable to the blower failure or the operational output reduction, or the blower start-up failure.

The latter is relevant to the partial load operation as reactivity failure. A start-up causes a throughput increase, which is not adapted to the secondary side throughput, and leads to the sinking of the hot gas temperature and to the increase of the cold gas temperature. This failure produces a reactivity transient and is recognized through the nucleation criteria

- Throughput ratio (primary to secondary side) above 1.3 and
- Cold gas temperature above 280 °C.

Thereafter, the RESA safety actions, blower shut-down and secondary circuit shut-off are initiated. The anomalous incidents are controlled through that.

As additional source of error, small deviations in the speed regulation, *e.g.*, through drifting of electronic components, which causes increase in speed and thereby associated increase in the output for constant outlet-gas temperature, influences the heat transfer ratio in the steam generator. Because of the high concentration in the steam generator, a significant live steam temperature rise can result from a small speed increase (2%), which can initiate exceeding of the of the design temperatures of components of the steam generator and of the live steam line. In order to counter it, an automatic live steam temperature limitation should be provided /U 2.6.7–1/, which limits the live steam temperature by lowering the outlet-gas temperature to 570°C, corresponding to a declaration of intent of the applicant. Should this measure be ineffective, the limiting device should initiate RESA and blower shut-down. With that, adequate safety is provided against malfunctioning of the blower speed regulation.

2.6.10.2 Failures due to Adjacent Systems

Failures, which are initiated by systems directly adjacent to the primary circuit and have effects on the primary circuit, are described below.

Nuclear Intermediate Cooling System KAB

The water cooler, which is arranged in the head space of the steam generator vessel, is supplied through the nuclear intermediate cooling system. There, this serves to cool the "stagnating Helium" to about 60°C in order to protect the operation of the blower with regard to the permitted operating temperature of the bearings and drives. A failure of the cooling system leads for a short period of time to the shut-down of the blower and with that of the RESA.

KBE Helium Purification facility

The Helium purification facility serves operationally to continuously supply a partial flow of the primary coolant over the corresponding cleaning equipments and lead it back into the primary circuit as purified.

In case of a breakdown of the purification line assigned to the unit, a switch-over to a reserve line is possible and is provided operationally. Should no other purification line be available, the complete failure of the Helium purification facility does not lead immediately to the limiting of the primary circuit operation. The further operation of the plant unit is possible only up until the specified permitted impurity values of the Helium are reached.

If the inflow and outflow of the Helium in the purified-gas store is also impaired by the failure of the Helium purification facility, with that the pressure regulation of the primary circuit also fails. The further operation of the plant unit is possible, as long as the operational range of the primary pressure is not crossed-over. The function of the pressure release system is not impaired by this.

Secondary Circulation

Failure in the live steam extraction or in the feedwater supply always causes a disproportion between secondary side and primary side output. Accordingly, the primary side is affected directly by secondary side failures through the thermal coupling until limitations or the reactor safety system become effective that lead to necessary safety actions.

The effects of the secondary side functional failures are analyzed and evaluated in Section 5.3 of this assessment. In addition, secondary side leakages and ruptures to the feedwater and live steam pipes are examined in Section 5.4.2. No unacceptable loads accrue from the evaluated course of failures for the components of the pressure-retaining enclosures.

2.6.10.3 External Influences

The primary circuit, inclusive of the primary circuit shut-off fittings, is planned to be designed for loads due to external influences. On the secondary side, this design should include the feedwater and live steam shut-off fittings. Hence, no loss of integrity and functionality results from EVA-load types due to the assumed destruction of the peripheral buildings and systems including the emergency power supply and the reactor safety system for the primary circuit.

The module facility is transferred in such a case to the "sub-critical, hot" condition. The afterheat dissipation from the core occurs through the extended surface cooler. Should the functioning of the cooling system also be impaired, the afterheat dissipation can be taken up again through the available fire service substation. This must take place within 15 hours after the occurrence of an anomalous incident to maintain admissible component temperatures.

2.6.10.4 Primary Circuit Exclusion

A safe shut-down possibility of the pipe lines connecting to the pressure vessel unit is a part of the safe primary gas inclusion. For this purpose, the pressure vessel unit fittings for the primary circuit shut-off (PKA) are arranged in suitable locations in the connection pipe lines. The PKA-fittings should be as close as possible to the pressure vessel unit yet be accessible during operation.

Therefore, the design provides

- Fuel element (BE-) charging and discharging installation FCA,
- Gas delivery system of the small sphere shut-down (KLAK-)-system JDP,
- Pressure compensation system JEY,
- Helium purification facility KBE,
- Oil supply of the primary circuit blower JEV, and
- Slotted measuring sections,

for the primary side connecting systems and pipe lines in the /U 2.6.10–1/ primary circuit shut-off system.

The PKA-fittings of the FCA-system are assembled at the end of the fuel element discharge pipe (rupture stripper block) in the form of fitting blocks and consist of an automatic PKA-fitting and a manual shut-off connected in series. They are arranged in the BE-discharge room. Just a manual shut-off is provided for the rupture cans since it is considered practically as a component of the pressure-retaining enclosure and has no continuing connections.

The automatic PKA-fittings with the manual shut-offs of the JDP-system connected in series are designed for the collective fitting blocks; these connect directly to the four nozzles in the bottom region of the reactor pressure vessel. Consequently, the fitting blocks are also arranged in the BE-discharge space.

The automatic PKA-fittings with their manual shut-offs of the KBE-connecting lines connected in series are included in the plan as outside the primary cell behind the wall bushing.

The JEY pressure compensation system has no automatic PKA-fitting, since the free passage of the conduit is required even for the primary circuit shut-off, if it is preceded by pressure gradients. The integration of the pressure compensation line to the EBE-extraction is therefore also located – seen from the primary circuit – with the PKA-fittings. This applies also for the branching of the JEG pressure release system, whose function must not be impaired by a primary circuit shut-off.

Furthermore, an automatic and a manual PKA-fitting are planned for each of the two connections of the blower oil supply, which are also arranged outside the primary cell. The emptying of the JEV-system sump in the blower does not require any automatic fittings since the manual shut-off provided is closed operationally.

All slotted measuring sections connected to the pressure-retaining enclosure are not included in the primary circuit shut-off in order to obtain structural information on the physical quantity of the anomalous incidents.

The control of the automatic PKA-fittings occurs for pressure release failures, provided the negative variable pressure gradient limit value is achieved. The performance requirements for PKA-fittings is that they lock based on elastic force for the maximum pressure difference of A_p=70 bar without auxiliary energy (failure-safe-principle). Further, they are designed to withstand loads from external effects.

This functional principle of the primary circuit shut-off requires that the PKA-fittings are kept open by auxiliary energy, *e.g.*, pneumatic. Thus for a shut-off requirement the open-status must be gradually shut-off. This requires an open-status system designed for EVA with redundant control. With that, the shut-off function is ensured even for EVA in each case. If one considers the failure of the auxiliary energy supply, the PKA-fittings shut off without delay even without reactor safety control. Hence, the PKA-fittings protect the primary gas inclusions

in case of anomalous incidents of pressure release. Thereby, the release of radioactive material is limited.

If the situation so demands, the primary circuit shut-off is not guaranteed at short notice on failure of a PKA-fitting. The additional possibility to shut off the manual shut-off connected in series can be considered as a medium term measure for the prevention of further formation of small leakages and for the core heating up phase after large ruptures.

Since a secondary system is necessary on the basis of the constructional execution of the automatic PKA-fittings for the operational open-status, a faulty shut-off of one or more PKA-fittings must be considered in case of the failure of the open-status system. This would lead to a non-availability of connecting systems, *e.g.*, the Helium purification facility. But a failure of the connecting systems does not lead to operational limitations of the plant in the short term. Short-term maintenance measures are possible, as the PKA-fittings are accessible during operation.

In summary, we consider the concept of providing only one automatic PKA-fitting and one manual shut-off fitting connected in series, which is located as close as possible to the pressure vessel unit but accessible from outside the primary cell, as sufficient.

On basis of the "fail safe"-execution of the fittings for the shut-off course, the primary circuit shut-off offers a fair reliability, even if only one automatic fitting per connection line is provided. Operating experiences exist for the fittings itself from the KVK-test bed, the THTR and the AVR /U 2.6.10–1/. Consequently, the primary circuit inclusion can be safeguarded by the primary circuit shut-off and the outcome of anomalous incidents as well as the radioactive spills can be limited.

Basically, unstoppable leaks for assumed ruptures of the primary circuit connecting lines between DBE and PKE-fittings or the pressure compensation pipe are also not to be excluded. These cases, despite being of low probability of occurrence, are considered as design failures and handled in Section 5.8.2 with regard to their radiological effects.

2.7 Reactor Auxiliary Installations

2.7.1 Helium Purification Facility/Helium Secondary Systems

Objectives

The Helium purification facility and the Helium secondary system have to fulfill following objectives as per the Safety Report:

- Cleaning of the primary circuits for commissioning before the initial start-up as well as after inspection and maintenance procedures,
- Purification of the primary coolant (Helium) by removal of dusty and gaseous contaminations within specified values,
- Removal of Tritium,
- Purification of the primary coolant by removal of additional radioactive contaminations, especially before their delivery to the purified gas-storage,
- Removal of water after an anomalous incident of water intrusion,
- Purification of freshly supplied Helium,
- Supply the Helium-filled systems with pure Helium,
- Pumping out the Helium from the primary circuit and the secondary circuits and bearings filled with Helium in the purified-gas store,
- Releasing pressure from or emptying Helium-filled auxiliary and secondary circuits and bearings containing radioactively contaminated Helium, depending on the requirement, and
- Evacuating the primary circulation and the Helium secondary system.

Layout, Design and Function

The Helium purification facility of the HTR two-unit power plant is built up from three cleaning conduits. One conduit each is allocated to each one of the two primary circuits. The third conduit is connected to the other two conduits; it is different from the other conduits because it also contains additional process installations for water separation.

The Helium purification facility and the auxiliary installations of the third cleaning conduit are dimensioned for an entry pressure of 1 to 69 bar and an entry pressure of 30 to 300°C.

Every cleaning conduit is dimensioned for a 5% gas change of the primary conduit per hour (135 kg/hour) for normal operation (250°C/60 bar). A run time of about 1000 hours is assumed until regeneration. The regeneration of the conduit is about 24 hours.

The auxiliary installations of the third cleaning conduit are dimensioned for a 100% gas exchange of the primary circulation per hour (3300 kg/hour) at 300°C and 60 bar for failure conditions.

Auxiliary systems are necessary for the operation of the Helium purification facility. Those exist only once for the three cleaning conduits. In particular, those are the following systems:

- Helium supply and storage,
- Pressure relief system for the Helium auxiliary circulation systems and BE handling.
- Evacuation systems for primary circulation and Helium purification facility, and
- Storage unit for radioactively contaminated Helium.

The layout and function of a cleaning conduit directly associated with a primary circulation is described in the following:

The extraction pipeline for the to-be-purified Helium is connected to a connecting piece of the steam generator pressure unit. The connecting piece is arranged in a way that the cold gas is extracted on the pressure side of the primary circulation blower. During normal operation of the cleaning conduit, the Helium is directed towards the two-tiered installations for extraction and retention of dust and gas contaminations. After recuperative reheating, the purified Helium is either directed towards the steam generator pressure unit or to the clean gas storage or to both at the same time with the aid of the Helium blower. The re-feeding connecting pieces in the steam generator pressure unit are arranged on the extraction side of the primary circulation blower.

The two stages of the cleaning conduit consist primarily of the following process installations:

1. Stage

- Dust filter
- Electrical heater
- Copper oxide bed
- Recuperative heat exchanger
- Water/Helium heat exchanger
- Water discharger
- Molecular sieve

2. Stage

- Nitrogen/Helium heat exchanger
- Low temperature adsorber
- Recuperative heat exchanger

In the first stage, the dusty contaminations are first retained in a dust filter (sinter metal filter). The temperature of the Helium can be in an electric heater heated to the operational temperature of the subsequent copper oxide bed, if necessary. There, the hydrogen (H_2) and carbon monoxide (CO) contaminations of the Helium are oxidized and are discharged in a recuperative heat exchanger and subsequent water/Helium heat exchanger in a molecular sieve after cooling off of the gas flow in the form of carbon dioxide and water (steam).

In the second step of the cleaning conduit, the Helium is again further cooled down to its operating temperature in a recuperative heat exchanger and in a subsequent Nitrogen (N_2)/Helium (He) heat exchanger, which is integrated in a low temperature adsorber. In the low temperature adsorber, residual contents of water and carbon dioxide are frozen out and split noble gases as well as the other gaseous contaminations are adsorbed onto the activated carbon. As described previously, the purified Helium can now be re-fed into the primary circulation or into the clean gas storage.

The design of the third cleaning conduit is different than that from the others. The difference of the otherwise likewise-built conduit is that there is a connecting pipeline between the integration of the conduit to the other two conduits at its intake and outflow, which carries a operation filter and an auxiliary installations for water discharge. The auxiliary installations consist of a failure cooler, a failure water discharger, and a failure blower. These auxiliary installations are started up after a water leakage into the primary circulation of one of the HTR modules. The cleaning conduit associated with that module is shut down in this case.

The function of the auxiliary installations is to remove the entered, vaporized water from the primary coolant in the primary circulation within a set time frame. The Helium-water mixture is cooled to about 50°C in the failure cooler and the condensed water is precipitated in the failure cooler and directed into the water extraction installation. The Helium is directed into the primary circulation with the failure blower. Before, a partial flow is diverted to the first stage of the third cleaning conduit, passes through different cleaning stages and arrives at the Helium supply and storage unit as clean gas.

The layout and function of the principal extraction and retention installations as well as the systems necessary for their regeneration are described in the following for normal operation of a cleaning conduit.

- Copper oxide bed

  The copper oxide bed is implemented as a solid state reactor (adsorption column) with two fills, where one fill consists of copper oxide and the other one consists of copper. The two catalysts are fixed on a tablet shaped supply material in highly disperse form /U 2.7–1/.

  The primary gas, heated to 250°C, is guided over the copper oxide bed. In this way, reductive contaminations are oxidized:

  $CuO + H_2 \rightarrow Cu + H_2O$
  $CuO + HT \rightarrow Cu + HTO$
  $CuO + CO \rightarrow Cu + CO_2$

If a break-through of hydrogen or carbon monoxide is determined, the copper oxide bed has to be regenerated. The copper oxide reduced by the contaminations is oxidized by oxygen during the cleaning process. A supply position for oxygen is located before the reactor bed. Oxygen is supplied until it breaks through and is detected by the oxygen probe located between the two fills. The excess oxygen is adsorbed by the copper fill. A pipeline filter retains the abrasion from the copper oxide bed.

– Molecular sieve

The molecular sieve is also realized as a fixed bed reactor (adsorption column). It contains a Zeolith fill as adsorption material /U 2.7–1/. The contaminations in the primary gas that were oxidized in the copper bed are adsorbed to the Zeolith as H_2O, HTO, and CO_2 after the gas flow has cooled down. For regeneration, the respective cleaning conduit is separated from the primary circulation and instead the third cleaning conduit is hooked up.

A shared regeneration system is provided for the regeneration of the molecular sieves of the three cleaning conduits, which consists primarily of the following components:

- Water/Helium heat exchanger,
- Water discharger,
- Auxiliary molecular sieve,
- Blower,
- Electrical heater.

During regeneration operations, the molecular sieve is sealed off, and streamed through counterclockwise to the normal operational flow and heated to about 300°C after pressure decrease. Thereby, carbon dioxide and water are driven out and the water is precipitated in the water discharger after the Helium has cooled down. The remainder content of water and the carbon dioxide are discharged in the auxiliary molecular sieve. Water and carbon dioxide are removed except for trace amounts during the subsequent evacuation of the molecular sieve.

The regeneration of the auxiliary molecular sieve occurs principally analogous to the molecular sieve by heating/purging, cooling and evacuation. The Tritium-containing waste water is discharged into the water discharger.

– Low temperature adsorber

The low temperature adsorber is realized as a low temperature-cooled fixed bed reactor (adsorption column) with integrated Helium/Nitrogen heat exchanger and vacuum jacket. An activated carbon fill is used as the adsorption material
/U 2.7–1/.

The primary gas, which was pre-cooled in the recuperative heat exchanger, is cooled down to the adsorption temperature in the Helium/Nitrogen heat exchanger before it flows through the activated carbon fill. Nitrogen, methane, Xenon, and Krypton are discharged there. The vacuum jacket serves as insulation.

For the regeneration of the three low temperature adsorbers, a shared regeneration system is also provided, which primarily consists of the following components:

- Water/Helium heat exchanger, Water discharger,
- Auxiliary molecular sieve,
- Blower,
- Electrical heater

The main function of the system is that the blower moves the Helium, warmed up in the heater, through the activated carbon fill, and thereby temperature-dependently vaporizes the frozen-out methane and carbon dioxide and discharges them. Subsequently, the low temperature adsorber is heated. Nitrogen is steam stripped and the other contaminations stored in the adsorber are also driven out and discharged into the pressure relief system.

In the following, the remaining Helium auxiliary systems that exist once each in the HTR two-module power plant facility are described.

- Pressure relief system for Helium auxiliary circulations and BE handling

This system follows to all systems from which Helium needs to be removed regularly during operation. It can accept all relief gases from the different systems simultaneously and pass them on to the Helium purification facility and the storage containers for radioactively contaminated Helium.

The main components of this system are:

- Relief container,
- Helium/water cooler,
- Buffer container,
- Compressor.

- Evacuation system for the primary circulation

  If required, the system serves the evacuation of the primary circulation and the associated Helium-filled auxiliary systems. The principal components of this system are:

  - Dust filter,
  - Pre-vacuum pump with exhaust filter (oil fog stripper), and
  - Main vacuum pump.

- Helium supply and storage

  This system supplies the reactor facility with the necessary Helium and can store the Helium pumping it from the Helium-filled systems. Together with the Helium purification facility, it serves the pressure maintenance in the primary circulation. The principal components of the system are:

  - Clean gas storage container,
  - Clean gas buffer container, and
  - Clean gas compressor.

- Storage of gaseous radioactive wastes

  This system serves the reception and storage of the radioactive gases that are generated during regeneration of the Helium purification unit. The system consists of two storage containers, whose supply occurs through the membrane compressor from the pressure relief system.

  The Helium purification facility and the Helium auxiliary systems are hooked up to the installation consumption supply. An exception to this are the primary circulation fixtures located in the loading and re-feeding pipeline, which are hooked up to the uninterrupted emergency supply.

For control and operation of the Helium purification facility and the Helium auxiliary systems, activity, pressure, temperature, and flow-through measurements are provided. Connections to the sampling system (gas analysis system) are provided at different locations for the control of the Helium contamination in the Helium purification facility and the Helium auxiliary system /U 2.7–1/.

In general and in case of a water leakage failure, the control of the Helium purification facility is performed manually.

The Helium purification facility and the Helium auxiliary system are not designed for external impacts. An exception to this are the sections of the loading and re-feeding pipeline from the pressure container unit to and including the primary circulation fixtures located in the reactor building UJA, which are designed for external impacts.

After shut-down of the cleaning conduit and the regeneration of the components that are contaminated with radioactive materials, entry for maintenance activities is possible. Maintenance activities at active components are not problematic since they can be performed with purified primary gas. Separate screens are provided for the dust filters, through which they can also be removed and handled.

Layout

According to the Safety Report, the Helium purification facility and the Helium auxiliary systems are located in the reactor auxiliary building UKA with the exceptions described in the following. The partial sections that serve the loading and re-feeding pipeline of the Helium purification facility, the evacuation system for the primary circulation, the Helium purification facility, and the storage container for condensate from the operation are housed in reactor building UJA.

The Helium purification facility is for the most part located between levels ±0.0 m and –9.5 m of the reactor auxiliary facility building. For the most part, a conduit-specific spatial separation is observed. Further, active and passive components will be located – jointly according to the aspect of expected frequency of operational access – in the separate rooms. The rooms, in which the control station and the gas analysis system are located, are arranged with respect to the other operational rooms of the Helium purification facility such that they have no open or locked connections to these rooms. The entire area of the Helium purification is separated from the other part of the auxiliary building.

The storage containers for Helium and liquid Nitrogen are located in the gas supply central unit UTG in the open area of the power plant. The connection to the reactor auxiliary facility building occurs through a pipeline and cable duct.

As assessment criteria for the assessment of the Helium purification unit and the Helium auxiliary system we have used the following:

- The guideline for radiation protection of personnel during maintenance work in power plants with light-water reactors "Provisions for Design of the Facility" from October 7, 1978 /L 15/, and
- Safety Criteria for Power Plants from October 22, 1977 /L 6/.

Based on the previously mentioned objectives, the Helium purification facility and the Helium auxiliary system are operational systems. The safety-technological importance is based on their inclusion and retention of radioactive materials.

The use of Helium systems in case of water leakage failure, on one hand serves for the reduction of humidity-based corrosion processes; on the other hand, it is supposed to prevent the activation of the primary circulation safety valve by limiting the pressure increase in the primary circuit. Because the activation of the safety valve is coupled with an activity release, this function of the system is a measure for minimization the activity releases into the surroundings of the facility. The function of the Helium purification facility is in this case not necessary for the failure control according to §28.3 of the StrlSchV /L 2/.

The Helium purification facility and the Helium auxiliary system are functionally designed according to their objectives. Positive experience from conventional and nuclear gas circulations is available regarding the functionality of the provided installations regarding their control technology with respect to retention and discharge of contaminations from the primary gas. These experiences are also confirmed through the operation of a Helium purification facility within the framework of a component test circulation for high temperature components /U 2.6.3–1/.

The proposed concept of the arrangement of the Helium purification facility and the Helium auxiliary system corresponds to their safety-technological importance. Because the Helium systems do not belong to the safety system, a strict redundancy separation is not necessary and the housing in a reactor auxiliary building that is not designed for loads from external impacts is appropriate. Due to the spatial separation of the passive component housing by conduit, those are accessible after regeneration for maintenance work. The partially shared housing of the active components does not impair their accessibility because they are installed in the cleaning conduits after the retention and discharge installations and thus are impinged by the primary gas.

The spatial and ventilation-technical separation of the entire area of the Helium purification from the other areas of the auxiliary facility building is an effective measure against activity carry-off and eventual leakages.

The clean gas storage containers for Helium in the gas supply central control in the outside area are to be classified as highly energetic containers. These containers should, according to the Safety Report, be arranged in a way that in case of a container detonation, no security-technologically relevant buildings can be hit by fly-away fragments. We consider such a layout to be realizable.

We consider the statements of the applicant regarding the layout of the Helium purification as being plausible and the intended layout of the discharge and retention installations including the auxiliary installations for the water discharge as being realistic.

Because the functioning of a Helium purification facility is not necessary for the failure control according to §28.3 StrlSchV the admissible maximum values of §28.3 will not be exceeded in case of a pipeline breakage of this system, a design of the Helium purification facility for loads resulting from external impacts is not necessary.

The layout of the loading and re-feeding pipeline from the pressure container unit up to and including the primary circuit seal-off fixtures against loads resulting from external impacts corresponds to the necessary requirements for the primary circuit seal-off.

The energy supply of the Helium purification facility as an operation system through the normal installation consumption energy supply is sufficient according to the safety-technological classification. The supply of the seal-off fixtures which belong to the primary circuit seal-off by the uninterruptible emergency energy supply is consistent with the requirements to be set forth for primary circuit seal-off.

We consider the designed instrumentation in the Helium purification facility and in the Helium auxiliary systems sufficient for the control and operation of the facilities. The instrumentation for measuring the contaminations in the primary gas is located within the sampling system (gas analysis system).

In case of a water leakage, switching to the third cleaning conduit of the Helium purification facility is supposed to occur manually. A timeframe of several hours is available in this case, thus, an automatic activation is not necessary.

The accessibility of the passive and active components is sufficient due to the spatial arrangement of the individual conduits and the gas conduit within the system.

In summary, we conclude that we consider the presented concept of the Helium purification facility and the Helium auxiliary systems to be suitable to fulfill the aforementioned objectives.

2.7.2 Sampling System (Gas Analysis System)

A gas analysis system is provided for the control of the Helium quality in the primary circuit that continuously measures the contamination of the primary gas. These chemical contaminations can be caused by air, water, or oil leakages that penetrate the primary circuit.

In addition to the operational objective of identifying chemical contaminations to control the Helium purification facility, the gas analysis system also has the safety-technological assignment to control the primary circuit for water leakages by measuring the humidity in the Helium gas with high sensitivity /U 2.5–5/.

In addition to the chemical contaminations, the gas analysis system also measures the radioactive contaminations.

The gas analysis system consists mainly of pipeline connections that are connected to different areas of the Helium purification facility. Thereby, the primary gas is re-circulated to the individual sampling devices through separate pipelines. The sampling devices work continuously under atmospheric pressure, which requires pressure reduction valves in the supply pipelines. Only the humidity measurement is carried out at system pressure to achieve the highest possible accuracy. Gas mice are only used where it is essential for technical measurement reasons (*e.g.*, Tritium, C14).

Because all measurements except for the gas chromatography are performed continuously, the measuring gas is re-circulated to avoid loss of primary circuit gas and gas discharges of the Helium purification facilities.

Two gas chromatography units, hygrometer, and oxygen meters are used as measurement devices. In addition, infrared devices, gamma-spectrometers, and a liquid scintillation spectrometer to measure Tritium are used.

The hygrometer is supplemented by infrared devices because a small water penetration rate of a few grams per hour cannot be safely determined by the hygrometer. The reaction product CO is detected by the infrared devices based on the water entry. In addition, a gas chromatography has to be carried out to discern whether the increase in CO is due to water entry or air entry.

Thus, only a discontinuous measurement is available for the detection of smallest water entries into the primary circuit. However, in cooperation with the continuous operation of the hygrometer, water leakages are detected with sufficient safety. The sampling connections for humidity measurements have to be located at the primary circuit in a way that the humidity control of failures of the Helium purification system adjacent to the primary circuit are not affected.

With respect to the required sampling devices and in addition to the state-of-the-art of analytical device technique, it should be paid attention to the fact that the analytical devices as well as the humidity measurement installations are operationally demonstrated.

The complete concept of the gas analysis is sufficient and suited to fulfill the objective of safety-technological humidity measurement of the primary gas and for the operational control of all the contaminations brought into the primary circuit.

2.7.3 Fuel Element Handling and Storage

2.7.3.1 Fuel Element Handling

Objective

A number of installations are provided for the handling of the sphere-shaped fuel elements of the HTR module, whose objectives are summarized in the following:

- Input and infiltration of new fuel elements into the reactor core,
- Removal of fuel elements from the BE discharge pipe,
- Circulation of partially burnt fuel elements,
- Discharge of BE breakage and non-conforming fuel elements,
- Discharge of burnt-off fuel elements and filling in transport and storage containers,
- Performance of the first core charging,
- As a particular case: the relocation of all fuel elements from the reactor into transport and storage containers and the re-charging of the reactor from these containers.

Daily, a portion of the fuel elements of each module is removed (about 5000), tested for damages, and tested for targeted burn-off. Of these 5000 fuel elements, after achieving the equilibrium state, about 360 fuel elements per full load day are directed into the transport and storage containers and replaced by fresh fuel elements /U 1/.

Layout, Design, and Arrangement

The fuel elements are transported in the pipelines either through gravity or pneumatically. About every 16 seconds, a fuel element is removed from and another one added to the core. The fuel element-advancing pipelines with their respective fixtures and functional parts are located in shielded duct and channels. The mechanical construction parts can be exchanged from accessible rooms. The burnt-out fuel elements are directed into transport or storage containers through pipelines.

The addition of fresh or partially burnt fuel elements to the core is carried out through the BE loading facility. The loading with fresh fuel elements occurs through a supply facility and supply sluice shared by both modules. The fuel elements roll individually from the transport container through the BE-separator into the pipelines. The fuel elements roll through a calibration device under atmospheric pressure and are collected in the supply sluice. The daily demand for a module is collected in this sluice section. Afterwards, the sluice section is locked, evacuated, and filled with pure Helium up to pressure equalization with the primary circuit. The new fuel elements are combined with the burnt-out fuel elements via a collection and conveyor block. From there, the fuel elements are transported up in a pipeline. They arrive at the core through a conveyor pipeline and a centrally located supply pipeline and in doing so are counted. The fuel elements are transported with primary coolant at cold gas temperature. It is intended that the drive-trains and functional units are located in accessible areas and can be controlled or exchanged.

The fuel elements are taken from the BE-discharge pipeline with the BE discharge facility. BE breakage and damaged fuel elements are discharged and moved to a so-called BE-jug. Depending on the burn-up results, the undamaged fuel elements are redirected into the supply unit or the discharge buffer. Burnt-up fuel elements are discharged and collected in a transport and storage container.

During the discharge process, the fuel elements arrive at two separators, which are coupled with a breakage discharger each through a breakage discharger. Only one separator/breakage discharger operates. The breakage discharger has a coil-shaped groove through which fuel elements with measurements not according to specifications and BE breakage fall into the breakage jug. Because the BE discharge occurs at primary circuit pressure, the breakage jug is designed as a pressure container; it is located in a mobile shielding container.

Undamaged fuel elements arrive at the burn-up measurement through a dosage-meter. If the burn-up condition is not yet attained, the fuel elements are transported back into the core.

Burnt-up fuel elements are collected in a sluice section. This sluice section is sealed off, pressure-relieved and filled with air or nitrogen during its filling with a maximum of 360 fuel elements. After pressure relief, the fuel elements are transported to the supply positions of the two transport and storage containers with 1.5 bar and room temperature. The filling of the storage containers is controlled by a counter. As soon as a container is filled, the next container is accessed and the loading block is undocked, and the container is prepared for discharge.

We used Safety Criterion 11.1 "Handling and Storage of Nuclear Fuel Materials and Other Radioactive Materials" and the BMI safety criteria /L 6/ as assessment criteria. Accordingly, installations have to be made, arranged and shielded in a way that a criticality failure and an inadmissible radiation exposure of personnel and the surroundings can be excluded.

The safety-technological requirements are therefore:

- Shielding,
- Observation of the burn-up value,
- Limitation of the outpouring cross-section (due to pressure relief failure)
- Design for EVA to appropriate extent,
- Observance of design limitations of fuel elements,
- Criticality safety.

In the presented planning of the components for fuel element handling, the applicant was able to refer to extensive experience for the predecessor facilities AVR and THTR-30 /U 2.7–2, U 2.7–3. Especially with the AVR, failures occurred relatively often in the fuel element transport systems. The reasons can be found in the not-yet-well engineered constructions and in the ancillary conditions, which only occurred during operation of the facility. Because the AVR was the first facility with spherical fuel elements, and, thus, with such a BE-supply facility, experiences had to be collected first.

The experiences gained from the failures resulted in improved construction of individual components for the THTR-300, which also led to improved operational results. From this occurrence, consequences were drawn for the THTR-300 that were also considered for he planning of the HTR module.

The conveying system is confirmed due to the operation in the said facilities and the preceding scientific examination.

Further, it can be expected that the design limitations for the mechanical load of the fuel elements can be observed through the improvements of the HTR module facilities for fuel element handling that the fraction of damaged fuel elements is reduced compared to the predecessor facilities. This is mainly due to elimination of absorber rods that enter the sphere fill.

The free fall of the fuel elements during their downward transport is reduced through the brake gas flowing against the fuel element. In case of a failure of the brake gas, the fuel element transport is suspended. The applicant has demonstrated that, even in this case, the design speed of 8.8 m/second is not exceeded for the fuel element still in the transport section /U 2.7–4/.

The operativeness of the construction parts and components and their sufficient accessibility has to be demonstrated in conjunction with analyses and tests. We do not foresee design determining aspects in this respect.

A sufficient shielding can be ensured due to the location of transport sections within channels. This ensures repair possibilities for wear parts even during reactor operation from accessible rooms.

The entire fuel element transport contains rotating dose devices and counters at different locations as well as a sampling device for controlling the burn-up. Thus, the fuel element transport is monitored continuously and it is guaranteed that the boundary limits of criticality safety are observed. Even in case of a transport pipeline this control guarantees that only a limited number of fuel elements can accumulate in the pipelines.

We consider the installations provided by the applicant as suitable to observe the previously-mentioned objectives and to fulfill the safety-technological requirements.

2.7.3.2 Fuel Element Storage

The storage of fuel elements in the HTR module is planned in two entirely different areas. On the one hand, fresh fuel elements are supplied for the use in the reactor, on the other hand burnt-up fuel elements are put in intermediate storage until final transport from the power plant site.

Layout, Design, and Arrangement

The fresh fuel elements are stored and transported in double-walled 200-L roll hoop barrels /U 1/. The gap between the outer and inner wall of the barrels is filled with finely grained ferroboron with a Boron content of 19 volume percent for safeguarding of undercriticality. One barrel has a capacity of about 1000 fuel elements. The storage for fresh fuel elements in the reactor auxiliary facility building ha a capacity of about 200 containers, which corresponds to a year's worth of fuel element supply. Hoisting and conveying devices are provided for the transport of barrels. A buffer storage for about 10 containers is located at the fuel element supply station.

Burnt-up fuel elements are filled in loose fill into a transport and storage container in the fill station.

After filling, the lid system is put on and a leakage and contamination test is performed. The container is transported to the reactor building crane with a rail-based industrial truck, lifted to from the mounting duct, and loaded onto a transport wagon for transportation to the intermediate storage area at the power plant. In the intermediate storage area, the containers are stored upright next to one another and connected to a container monitoring system.

For an assessment of the described fuel element storage, we have used the BMI Safety Criterion 11.1. "Handling and Storage of Fuel Elements and Other Radioactive Materials." The following design requirements for the storage of burnt-up fuel elements that need to be complied with during normal operation and during failures follow:

- Safe observance of undercriticality

  The containers are to be designed in a way that the development of a critical arrangement during handling, storage intended by design, and all assumed failures is impossible.

- Safe discharge of fission heat

  With respect to heat technical characteristics, the BE containers are to be designed in a way that the fission heat is always safely discharged and the temperatures of the BE and the containers stays below the respective admissible values.

- Safe inclusion of radioactive materials

  The loaded containers need to be executed in a way that during the intended operation no radioactive materials are released from the containers and that their leak-tightness can be dependably monitored. Containers and container seal-offs have to keep their integrity even during loads from external impacts.

- Shielding of the ionizing radiation

  The burnt-off fuel elements are typically handled and stored in containers that correspond to those used for burnt-off LWR fuel elements. The storage is comparable with that of the planned storage of AVR and THTR fuel elements in the AVR container storage facility in Jülich or the container storage facility Ahaus. We have examined the principal suitability of the containers and the type of storage in our assessments of the AVR container storage facility in Jülich /L 78/ and the container storage facility in Ahaus /L 79/. We concluded in these assessments that there are no objections to the planned storage in the two mentioned storage facilities. The handling of burnt-off fuel elements in the HTR module and the storage in the intermediate storage facility on the premises of the power plant is comparable to that of the mentioned storage facilities. For this reason, we are of the opinion that the provided installations regarding the fuel element storage are suitable to fulfill the objectives under the said conditions.

The containers for transport and storage of fresh fuel elements had already been used for the THTR-300. They are suitable for the proposed objectives to fulfill the set requirements. For failure situations, we refer to Section 5.5.2 of this assessment.

2.7.4 Facilities for the Treatment and Storage of Radioactive Wastes

2.7.4.1 Wastewater

Objective

Wastewater from the reactor facility and the reactor auxiliary facility buildings that is radioactively contaminated due to the operational processes is collected in the facility for the treatment and storage of radioactive wastewater, and, depending on its condition, decontaminated.

The requirements for the facility arise from the requirements for safe inclusion of radioactive materials, the minimization of activity release, and the obstruction of uncontrolled discharge of wastewater.

Layout, Design and Function

The radioactive wastewater treatment and storage facility is installed in the auxiliary reactor building. The individual features of the design are given in the Safety Report.

Wastewater is divided into two categories based on their radioactivity and origin (category I: radioactive wastewater; category II: minimally radioactive to inactive wastewater) and diverted to the respective wastewater collecting containers. Two procedures are designed for the following decontamination process:

- Evaporation, and
- Cleaning by a centrifuge facility.

The facility is constructed in a away so that all the steps of the process can be performed in any combination or repeated, if required. Decontaminated wastewater is collected in the control containers. If a decision sampling determines that the activity concentration is lower than 1.85 x 10^7 Bq/m^3, it is possible to discharge it under control.

The system of radioactive wastewater treatment and storage consists of the following components:

- 4 Wastewater collection containers,
- 1 Evaporation facility,
- 3 Control containers,
- 2 Concentrate containers,
- 1 Chemical distributor.

Wastewater is collected for further treatment in the four wastewater collection containers that have a net capacity of 20 m^3 each. Two containers are designed for the wastewater of each group – *i.e.* category I and category II. Wastewater in containers is prepared by adding chemical agents for further treatment. Mixing and sampling in the containers is possible.

The remaining contaminants and eventually minimally active components of category II wastewater can be cleaned in the centrifuge facility. Category I wastewater can be pre-cleaned here prior to evaporation. Fine sludge periodically discharged from the separator is temporarily stored in the sludge container and is later dosed to the decanter feeding water. Residual solid materials from the decanter are fed directly to the wastewater barrel. The rated flow rate of the centrifugal facility is 0.7 kg/second.

Separation of the radio-nuclides from contaminated wastewater is carried out by evaporation. The facility consists of an auxiliary steam-heated evaporator with natural circulation, a pipe bundle, and a vapor vessel – each of them includes a separating tank, condensate, gas cooler, gas discharge and distillate cooler. Cooled distillate is fed to the control container and the remaining evaporation concentrate is periodically discharged to the concentrate container. The rated flow of the evaporation facility is 0.28 kg/second. The decontamination factor is given as 10^3 up to 10^5.

Cleaned wastewater is collected and homogenized in the three control containers; each of them has a net capacity of 20 m^3. Finally it can be discharged by the discharging facility in a controlled manner, after having undergone final measuring. It is possible to feed insufficiently decontaminated wastewater for subsequent cleaning or to return it to the wastewater collection container.

Concentrate from the evaporation facility and separated sludge from the wastewater collecting containers are fed to the two concentrate containers, which each have a net capacity of 10 m^3. Concentrates can be stirred by a motor-driven device and kept in suspension in the containers. The sludge pumps discharge the concentrate for further treatment.

The evaluation of the wastewater system is based on the following decisive criteria:

- Safety Criteria for Nuclear Power Plants /L 6/,
  Safety Criterion 10.2: Activity Monitoring in Wastewater
- KTA rule 3603 /L 50/,
- Guidelines for Discharging Wastewater from Nuclear Power Plants with Light-water-Reactors to Waterways (LAWA) /L 24/,
- KTA rule 1504 /L 64/.

We have examined during the evaluation of the water discharge systems whether the designed facility can fulfill the tasks given in the description and whether it is designed in compliance with the evaluation criteria. With regard to the storage and treatment capacities we have taken into account the fact that this is a considerably smaller facility than existing light-water-reactors.

We have examined whether:

- The designed treatment procedures as a whole presents a appropriate concept for the treatment of wastewater resulting from the designed operation,
- The design of the wastewater treatment and storage capacities is sufficient for the given quantity of wastewater,
- The appropriate measures according to the state-of-the-art of science and technology are planned or necessary in the case of an uncontrolled release of radioactive materials to the surroundings of the power plant after anomalous incidents or failure.

The designed treatment concept of radioactive contaminated wastewater treatment as presented in the Safety Report was selected based on extensive experience with the operation of such facilities.

The entire facility is designed so that individual procedures can be used based on the requirements resulting from various wastewater quality. Site-dependent details can be met by the design of the facility with respect to cleaning grade, pH-Value, and load by dispersed materials, etc.

Operational experience with the wastewater treatment facilities in nuclear power plants with light-water reactors has shown that the designed capacity is sufficient for the facility in the HTR-Module, which should decontaminate a yearly quantity of wastewater as required by KTA rule 1504 and KTA rule 3603. The designed storage container provides a buffer capacity, which makes it possible to collect and handle a peak quantity of 40 m^3/day. Flexible operational control is possible based on the designed treatment methods as well as the treatment and operation buffer capacity, which makes it possible to observe the required maximum limits for the activity concentration of 1.85×10^7 Bq/m^3.

To prevent exceeding the maximum concentration of 1.85×10^7 Bq/m^3, as required by KTA rule 1504, an automatically operating shutoff system has been designed for the output line. This shutoff system is designed to protect against functional failure so that uncontrolled active wastewater cannot be transferred to the sewage system as a result of failed operational conditions or a failure. For this purpose, the respective equipment is designed to fulfill Safety Criterion 10.2.

In the case of a long-term failure of the wastewater treatment systems, the function of the connected systems and of the entire facility is affected. As the system has no safety-technological function, the requirements beyond those given in the description and the system operation do not apply to the function of safety-technologically important systems, and our task is to evaluate only operational failures, in which radioactive materials are released.

We examined the following internal failures of the facility:

- Power supply failure,
- Failure of the auxiliary steam supply,
- Failure of cooling water supply, and
- Loss of component integrity and function.

The results of these investigations show that no failure results in effects that would necessitate a design change. Technical measures for failure control are possible and some of them are already included in the concept. The method and scope of such measures is given in detail in the implementation plan.

Higher radiological effects occur as a result of leakage from wastewater concentrate containers. We refer to Section 5.8 of this assessment for the detailed examination of failure consequences.

The wastewater treatment system is not protected against induced vibrations caused by external impacts. The radiological effects of a release of radioactive materials resulting from an earthquake are also given in Section 5.8.

If external impacts occur, we cannot assume the further operation of the power plant at least for an interim period. In our opinion there is a long-term possibility to collect and treat the wastewater in an external facility. There are no operational reasons for the protection of the wastewater treatment system against external impacts.

In our opinion, the facility for radioactive wastewater handling and storage is designed in a way that is appropriate for the treatment of all radioactively contaminated wastewater resulting from the operation consistent with the design, and all requirements can be fulfilled. This provides a prerequisite according to the state-of-the art of science and technology for ensuring considerably lower values than required during an average wastewater discharge and during the appropriate control of operational procedures. The design documents provide for technology which, in our opinion is able, to prevent uncontrolled wastewater discharge into the sewage system. These treatment facilities comply with the state-of-the-art of science and technology. As the design complies with the objective and with the requirements put on the designed facility, we have no objections to the presented concept.

2.7.4.2 Solid Material Storage

According to the Safety Report, the following solid waste materials must be considered:

- The concentrates produced,
- Consumable parts of the nuclear-technological facilities,
- Filtering tanks from the decontamination systems,
- Consumable filters from the nuclear-technological ventilation facility and other systems of the control area,
- Consumable equipment, covers, laboratory waste, etc.,
- Decanter residuals.

The waste is collected in containers or in 200 L-roll hoop barrels made and stored there until they are conditioned in the power plant. After external or internal conditioning by mobile facilities, the waste is again stored in barrels in the interim barrel storage.

In our opinion, the capacity of the barrel storage (200 L barrels) is sufficient dimensioned to hold at least the waste produced within two years. We consider the designed capacity as sufficient. In addition, other interim storages can be obtained.

Heat development based on the activity inventory is assumed to be so small that no additional measures for heat removal are required. Removal of airborne radioactive materials, which may eventually be released from the barrels, occurs in a controlled way. For these reasons we have no objections to the type of interim storage. As for the fire protection aspects, our statement is given in the assessment on fire protection.

2.7.4.3 Decontamination Facilities

The decontamination area in the auxiliary reactor facility building is available for the decontamination of machine parts, fixtures, tools, etc. Smaller parts are cleaned manually in the washing basins using acids, lyes, and warm or hot water. Larger contaminated parts are also treated in the decontamination area using the special decontamination box – by water, steam or acids. Parts, whose minor surface wear is not critical, may also be cleaned in the high water pressure lapping facility. The decontamination facilities are equipped with air suction pumps.

The mobile decontamination system is designed for cleaning the inner walls of the auxiliary and accessory facilities. The system is installed on two mobile stands and in general consists of chemical containers, a high-pressure cleaning pump, a dosing pump, an exhaust pump, a heating device, a residual filter and the required connecting lines. According to the type of contamination, decontamination by high-pressure water or with chemical cleaning agents is carried out.

The presented decontamination principle makes it possible to select the appropriate procedure as well as the appropriate decontamination agent for various requirements given by the different materials and contaminations. The concept complies with the state-of-the-art of science and technology. We agree to the application of the presented procedures.

2.7.5 Nuclear Building Water Discharge System

The function of the nuclear building water discharge system is to collect any waste liquids from the control area that may be contaminated and to feed them to the system for treatment and storage of radioactive wastewater.

According to the Safety Report, the building water discharge system includes partial systems for water discharge from the premises

- Inside the reactor building,
- Inside the reactor building extension, and
- Inside the reactor auxiliary facility building.

In the various building sections waste liquids flow along floor outlets, pipelines and interim reservoirs (sumps) to the pump sumps by the force of gravity. The wastewater is pumped from the pump sumps into the system for radioactive wastewater storage.

The design of the building water discharge systems is based on the respective quantities of waste liquid as well as on their activity content. The resulting requirements for shielding as well as for dimensioning and selection of materials for the system can be solved within the framework of the implementation plan. There are no objections to the concept of the nuclear building water discharge system.

2.8 Ventilation Technological Facilities

While evaluating the ventilation technological facilities of the HTR two-module power plant facility we started with the standards and regulations set for ventilation technology and KTA rule 3601 "Ventilation-technological Facilities in Nuclear Power Plants" /L 47/. In addition, we used our own experience with the construction and implementation of ventilation facilities in other nuclear-technological facilities.

2.8.1 Nuclear-technological Ventilation Facilities for the Control Area

Functions

According to the Safety Report /U 1/ the nuclear technological ventilation facilities have to fulfill the following functions:

- Supply of the buildings with fresh air from the outside,
- Observance of the specified air condition inside the building,
- Observance of negative pressure and directions of ventilation (air flows),
- Discharge of heat loss from mechanical and electrical components,
- Discharge of radioactive gases and atmospheric particles from the ambient air through rinsing and filtering,
- Minimization of contamination of the surroundings by targeted and filtered discharge of waste air, and
- Pressure relief in the reactor building or in the Helium lines in the auxiliary reactor facility building, respectively, through the discharge valve and the transfer of the pressure to the chimney in the case of a failure in the pressure relief system.

To be able to maintain sufficient negative pressure in the rooms with higher contamination potential, the following areas in the control area room are separated within the ventilation technology system:

Reactor building:

- Primary cells,
- Facility and operational premises,
- Supply section and reactor hall as well as the reactor building extension (section of the interim cooling systems).

Auxiliary reactor facility building:

- Facility rooms (Helium section),
- Facility rooms (water section),
- Laboratory and sanitary rooms.

KTA-Regulation 3601 includes many general requirements, which apply to the concept of the ventilation technological facilities in the nuclear power plants:

- Air ventilation from the control areas should be controlled so that waste air is ventilated off only through the intended conduits.

- Air from the exhaust air conduits must be filtered so that radioactive Iodine and radioactive atmospheric particles are captured, as far as this is necessary for the protection of the surroundings according to the design layout of the nuclear power plants facility.

- Iodine and atmospheric particle activity concentrations in the ambient air should be reduced to the minimum possible level by ambient air filtering facilities or by measuring the waste air quantity, as far as this is required for protection of the staff.

- Radiation protection-technologically required negative pressures in rooms, negative pressure gradation in rooms, and air flow directions must be observed.
- Specified admissible inside air temperatures, air humidity, and the minimum air exchange must be observed.
- Accessible rooms must be supplied with the required amount of fresh air.
- Inflammable and harmful gases and vapors must be removed with the ambient air.

A comparison with the requirements set forth for the ventilation facilities for the HTR module mentioned in the Safety Report shows that the concept of these ventilation technological facilities fully complies with the requirements given in KTA rule 3601. In addition, the technological design of the nuclear-technological ventilation facilities system is described from the point of view of compliance with the specific safety-technological requirements specified for the design of the HTR module.

Layout, Design and Arrangement

The ventilation technological equipment in the control area consists of the following main components:

- Outside air facility,
- Exhaust air facility,
- Secured control of negative pressure by active filters and atmospheric particulate filters,

- Exhaust air filtering facility for filtering partial air flows,
- Ambient air cooling facilities for the facility and operation rooms in the module area,
- Ambient air cooling facilities in the fuel element pipeline and the fuel element exhaust area, as well as
- Pressure relief valves in the chimney.

The schematic design of the ventilation facilities for the reactor and auxiliary reactor facility buildings is given in the Safety Report. We have presented the designed data of the important nuclear-technological ventilation facilities in Table 2.8–1.

Except for the ambient air ventilation facilities, which are situated directly before the facility on site and the operation area in the HTR module, all main components such as filters and blowers are contained within in the auxiliary reactor facility building.

Classification of the nuclear-technological ventilation facilities in the above mentioned sub-area is in compliance with the specific requirements set forth for HTR module power plants. The construction and the process-technological design of the individual ventilation facilities fulfill the safety-technological requirements set forth for the individual sub-areas in the Safety Report. In our assessment we have given particular importance to the subject of an adequate redundancy grade from the standpoint of the safe function in case of failure. We have described this topic in more detail in sections 2.8.1.3 and 2.8.1.4. We conclude that the design also fulfills the set requirements in this aspect.

Table 2.8–1 Design data of the nuclear technological air ventilation facilities

Outside exhaust airfacility/ambient exhaust airfacility

Blowers	2 × 100% each
Air flow volume	30.5 m^3/second
Power supply	normal grid
Cold water supply	conventional cold water system (outside exhaust air facility)

Ambient air filtering facility

Blowers	1 × 100%
Air flow volume	5.55 m^3/second
Power supply	normal grid
Cold water supply	2 × 50% class S

Safe negative pressure maintenance

Blowers	2 × 100%
Air flow volume	2.5 m^3/second
Power supply	normal grid
Cold water supply	1 × 100% Class S 2
Activated carbon filter	1 × 100%

Circulating air facilities

Localized circulating air facilities are provided for in rooms of the control area with higher heat generation.

Ventilators	100% each
Power supply	normal grid
Cold water supply	conventional cold water system

We have no objection to the concept of the layout of the nuclear-technological air ventilation facilities. It complies with the state-of-the-art of science and technology. However, we point out the fact that lower local dosage must be designed in the implementation plan of the layout of ambient air ventilation facilities on the premises.

Functional Consistency during the Designed Operation

During the designed operation of the power plant facilities, fresh air supply, air circulation, and exhaust air facilities are continuously operated whereas the exhaust air facility and the secured negative pressure control unit are in standby status. Any negative pressure in the various facility areas is controlled by the air supply valves in the corresponding air supply channels.

If required, the air can be vented from the reactor building and auxiliary reactor facility building to the chimney of the exhaust air filter facility. This may be needed particularly in all the areas of the reactor building as well as for the facilities of the auxiliary facility building, except for the sanitary and laboratory facilities. Except for the air from the facilities of the water processing section, the air flows from these areas can also be directed through the secured negative pressure control unit with an activated carbon filter.

The negative pressure control in the facility and directed airflow control by the air supply control complies with the state-of-the-art of the technology used for fulfilling comparable requirements. Redundancy discharge of active components in the air supply and ventilation facilities as well as secured negative pressure control provides adequate availability. In our opinion, the only need is to provide the possibility of filtering the air vented from the laboratory rooms according to the requirements of DIN 25425, or DIN 25466, respectively.

The values of radioactive materials vented with the exhaust air expected during the normal operation of the HTR two-module power plant facilities are listed in Section 3.4.1. Based on these values it is admissible to discharge exhaust air without filtering during normal operation and so the exhaust air facility can be left out-of-operation.

The exhaust air facility must be operated during repairs or revisions, as it is necessary to retain the atmospheric particles that may occur. In the case of a failure of this connected facility, there is the possibility of feeding the exhaust air through a filter in the secured negative pressure control unit.

In the following section we present our opinion on the functioning of the secured negative pressure control unit, which is designed to minimize activity release and negative pressure control after a failure.

Function in Case of Failure

In this section we have examined whether the designed safety-technological requirements for the designed failure control can be fulfilled by the nuclear-technological ventilation facilities, particularly pertaining to the need for a emergency power supply and redundancy grade.

– Need for a emergency power supply

Out of the ventilation facilities in the control area, only the secured negative pressure control unit is supplied by a emergency power supply. Pressure level control in the case of the need of a emergency power supply is implemented by the secured negative pressure control unit. In general, exhaust air is thus filtered.

The controlled value of the air vented to the chimney is ensured by the negative pressure control in the building. The active components of the secured negative pressure control are designed as redundant. A subjected failure of the filtering facility – which is provided as an individual facility – has no effects on the need for a emergency power supply, as exhaust air is not filtered during normal operation and the need for a emergency power supply during normal operation does not lead to any increased values of radioactive materials.

Ventilation facilities designed for cooling are not supplied with a emergency power supply, which is why they stop operating for the time period of supply from a stand-by power supply source. It must be shown in the implementation plan that no inadmissible temperatures could occur here.

- Failure of the Pressure Relief Control

Failure of the cooling material system was analyzed in Section 5.4.1. Large leaks that lead to rapid pressure reduction are differentiated from the occurrence of leaks that result in small losses of coolant. The safety-technological function of the nuclear-technological ventilation facilities must be designed for the entire spectrum of leaks. In the following, we discuss the requirements of different typical cases of different leakages of various sizes.

- Large leaks (breaking away of any of the DN 65 connecting lines)

  Excess pressure that occurs during such a failure is removed by ventilation to the chimney – air is fed off through the over-flow opening inside the reactor building and then through the pressure relief valves which open at an excess pressure of 10 kPa at the latest. After the pressure balance is restored the valves are closed automatically in order to enable restoration of negative pressure control. If the closing function fails, it is possible to manually control the air ventilation valves, which are installed ahead of the pressure relief valves.

An appropriate pressure relief system is also provided for the facility rooms (Helium line) in the auxiliary reactor facility building. Pressure increase by the channels in the adjacent rooms is prevented by the built-in set-back flap valve.

The required pressure level control is restored by secured negative pressure control after the pressure relief valves are closed. If no automatic action occurs, it is possible to switch on the secured negative pressure control manually. There is a sufficient time period for doing so.

In the following core heating-up phase, the exhaust air is removed through atmospheric particulate and activated carbon filters in the secured negative pressure control system. In the case of a subjected failure of this filter system, the exhaust air can be fed through the atmospheric particle filters of the exhaust air facility.

In our opinion the concept of different methods of ventilation in the two subsequent phases of pressure relief failures and the design of the nuclear-technological ventilation facilities based on the amount of discharge of radioactive material discussed in Section 5.8.2, is acceptable, and adequate minimization of radiation exposure into the surroundings is ensured.

- Small leakages (breaking off of a measuring line)

  In the case of a broken measuring pipeline (DN≤10) or a failure of a small safety valve, the air supply is reduced (throttled down) and switched over to the secured negative pressure control. The failure criterion here is achieving the activity limit value in the exhaust air.

  In our opinion the measuring-technological design, the provided design measures and the design of the nuclear-technological ventilation system are acceptable – they are capable of ensuring adequate minimization of radiation exposure of the surroundings in the case of small leaks in the primary circuit. This also applies to the case, where automatic switch-over failed and the secured negative pressure control must be switched on manually.

  The release of radioactive materials for such a case is described in Section 5.8.1.3; an analogous procedure must be performed in the case of big leaks if an additional failure of Iodine filtering is assumed.

- Medium-size leakage

  Leakages can occur, where on one hand, they are not yet followed by the activation of the pressure relief valves, but on the other hand a pressure increase in the building occurs through with the possibility of a discharge along the floor through building leakages. This may occur if the leakage is larger than the supply amount of the secured negative pressure control system.

  These issues are discussed in more detail in Chapter 2.4.2 "Safety Enclosure".

-- External Impacts

The reactor building is protected against external impacts. The ventilation-technological facilities are not protected against induced vibrations resulting from external impacts.

Poisonous or explosive gases can be kept from the facility by closing the valves in the air supply channels. Dangerous explosive gases are detected by the gas warning system.

> Due to the design of the systems of the HTR two-module power plant facility as protected against induced vibrations resulting from external impacts (EVA), the ventilation facilities are not designed for activity retention after EVA. For these functional reasons a corresponding design of the nuclear-technological ventilation facilities is not required. Within the framework of the implementation plans of the system account must be taken of a failure of individual ventilation components caused by external impacts so that they do not influence the function of other systems required for failure control. Eventually such ventilation components should be designed as protected against induced vibrations caused by external impacts according to their security level.

2.8.2 Ventilation Facilities for Non-nuclear Operation

Functions

Of the ventilation facilities outside of the control area, only those, which are important from a ventilation-technological aspect, are discussed here. These are the ventilation-technological systems for the switching stations and emergency supply facility buildings and for the emergency control point in the reactor building. All other ventilation systems are not of safety-technological importance, and are designed and implemented according to the general regulations.

In addition to supply the buildings with fresh air and controlling the specific temperature and humidity values of the room air, the safety-technological function of ventilation is specified as the removal of heat removal from cables and electrical facilities. In addition to this, the facilities remove gases from the battery rooms and are used for directed fume removal in case of a fire.

2.8.2.1 Ventilation in the switching stations and emergency power facilities

Layout, Design and Arrangement

The individual ventilation systems in the switching stations and emergency power facilities and their designs are summarized in Table 2.8–2. The air-conditioning systems in the maintenance area and the ventilation blowers for the battery rooms are supplied from a stand-by power supply source, if necessary. The air-supply and exhaust systems are equipped with coolers, heaters, and dehumidifiers, and input air filters as required for their function.

The facilities are installed in the switching stations and emergency supply buildings situated at +10.90 m.

The facilities are planned, designed, arranged, and supplied with power according to their safety-technological function.

Function during Designed Operation

The air supply system provides fresh air for all the buildings in individual building sections. They are designed and partially connected to the ventilation facilities for both switching boards and air-conditioning system for the working area and electronic rooms; other rooms are supplied directly.

The exhaust air facility takes in the air from air supply facilities in the room area and then discharges it to the outside.

The ventilation facilities for switching stations as well as for working areas and electronic rooms operate with a constant supply of fresh air and also secure the removal of heat from these areas.

The designed function of the ventilation-technological equipment corresponds to the specified requirements and the requirements placed on such equipment.

Table 2.8–2 Critical data of ventilation in the switching station and emergency supply buildings

Outside exhaust airfacility/ambient exhaust airfacility	
Blowers	2 × 100%
Air flow volume	10.27 m^3/second
Power supply	normal grid
Cold water supply	conventional cold water system
Filter	category EU 4 (DIN 24815)
Exhaust air facility	
Blowers	2 × 100%
Air flow volume	10.27 m^3/second
Power supply	normal grid
Ambient air facilities (2×)	
Blowers	1 × 100% each
Air flow volume	10.27 m^3/second
Cold water supply	conventional cold water system
Power supply	normal grid
Air-conditioning control room, computer and electronic rooms	
Blowers	2 × 100%
Air flow volume	15.34 m^3/second
Cold water supply	Secured conventional cold water system
Power supply	Emergency power supply
Activated carbon filter	Category EU 7 (DIN 24185)

Exhaust air battery rooms (2×)

Blowers	1 × 100%
Air flow volume	0.37 m^3/second
Power supply	Emergency power supply

Because of the linked structure of the facilities, the air supply and ventilation facilities can take over the cooling of the building structure in the case of a ventilation facility failure. The air-conditioning facilities for the control area and electronic rooms and the remaining air circulating facilities then work under a clean air circulation operation. Even in the case of a failure of the air-conditioning facility for the control area and electronic rooms, the appropriate switching of ventilation operation can be observed under a slightly increased temperature.

Appropriate provisions are provided for the protection against internal facility failures through the concept of the facility as well as its redundant structure.

Function during Failures

– Emergency power supply case

During emergency power supply operation, the air supply, ventilation and air circulation facilities are not supplied with power. The diesel fuel sets can then directly release heat via the fresh air-cooled systems. Cooling of the control areas and electronic rooms and ventilation of the battery rooms is observed.

In case of emergency power supply, cooling of the control area and the electronic rooms is reliably ensured by the redundant design of the active components of the air-conditioning facility and by connection to the redundant secured cooling water system which are both supplied by the emergency power source.

In case of emergency power supply, ventilation of the battery rooms is provided in the same way as during normal operation.

The ventilation facilities of the switching station and emergency supply building that are not supplied by the emergency power supply are generally not required in case of emergency power supply from a safety-technological standpoint. However, it must be shown within the framework of the implementation plan that no inadmissible temperatures are reached in the switching stations during emergency power supply in the case of a failure of these facilities.

– External impacts

The switching station and emergency supply building as well as the safety-technologically important facilities installed in it, *e.g.*, the reactor protection system and emergency power supply facilities, including auxiliary systems, are designed for protection against earthquake. Operation staff can monitor the facility, test, and control the condition of the facility. By this the heat loss is distributed to the air in the rooms. The ventilation systems for the switching station and emergency supply building are not designed as protected against earthquake.

We cannot accept without reservation the applicant's plan that the ventilation systems in the switching station and emergency supply building are designed as generally not protected against induced vibrations caused by earthquake. In our opinion it must be shown within the framework of the implementation plan that no inadmissible conditions could occur in the facilities and systems affected by an earthquake and with which measures such conditions can be prevented.

The design of the ventilation systems as protected against induced vibrations caused by not very probable impacts from aircraft attack and explosion waves – the same, as the designed concept of the building – is not required. The control of this situation applies only to the facilities of the designed reactor building that is protected against any external impact.

2.8.2.2 Ventilation of the Emergency Control Point in the Reactor Building

Layout, design, and arrangement

The ventilation systems of the emergency control point and their design are given in the following summary.

<u>Air-conditioning – emergency control point</u>

Blowers (air intake/discharge)	2 × 100 % each
Flow rate	0.55 m^3/second each
Power supply	1 × emergency power supply each
	1 x battery power supply each
Cold water supply	Conventional cold water system

<u>Ventilation of the battery room – emergency control point</u>

Blower	1 × 100 %
Flow rate	0.16 m^3/second
Power supply	Emergency power supply

The ventilation systems for the emergency control points are installed adjacent to the emergency control system in the reactor building, which is protected against all external impacts. The pressure compensation systems in the air input opening protect the ventilation-technological facilities of the emergency control point against a shock wave.

The emergency control point is separated from the reactor building ventilation system by the air-conditioning facility, which is supplied with filtered, cooled or heated and humidified or dehumidified fresh air. Coolers are supplied with cold water by a conventional cold water system.

The structure and layout of the ventilation systems comply with the designed objectives of the systems and with their safety-technological importance. The design of the ventilation system reflects the fact that the emergency control point is required only in rare and exceptional situations and the control requirements occur in addition to the action of the KLAK-close-down system.

Function during Designed Operation

Air can be supplied or discharged to/from the emergency control point by air-conditioning and ventilation systems during the designed operation. Lost heat is removed either by the ventilation or air circulating system. Ventilation of the battery room is fed outside by a separate ventilation blower. This blower is switched off under individual air circulation operation in the emergency control point.

The ventilation systems in the emergency control point are ready to ensure the requirements of the control of the room air condition as well as securing the operational readiness of the emergency control point and its auxiliary facilities. The requirements are met during the designed operation of the HTR two-module power plant. There are no other requirements placed on these ventilation systems during the designed operation.

Function in Case of Failure

– Emergency power supply case

The ventilation systems of the emergency control point, which are separated from the other ventilation-technological systems of the reactor building, are supplied with power from an emergency power supply diesel generator. One of the two installed air supply and discharging blowers is always connected to the independent battery-supported emergency power supply system of the emergency control point. The conventional cold water system is not available for cooling in the case of an emergency power supply

In the case of power supply from the emergency power supply diesel generator, the power plant control must operate in a way that does not require emergency control point. The operational readiness of the emergency control point is ensured by the emergency power supply of the ventilation systems in the emergency control point.

It must be specified within the framework of the implementation plan, whether sufficient heat removal from the emergency control point is ensured in the case of a failure of the cold water supply of a cooler.

In the case of the failure, "Need for an Emergency Power Supply with the Failure of the Emergency Power Supply Diesel Generator" the required monitoring functions are provided by the emergency control point after which control is no longer possible because of the limited battery capacity, *i.e.* after two hours. With the designed battery capacity, the ventilation function can be operated for 15 hours in total.

– External impacts

After an anomalous incident caused by an aircraft attack or a shock wave, control loosening can be expected. The required monitoring of the facility will be provided by the emergency control point. This also applies in the case of an earthquake if the regular main power supply fails for more than two hours and the stand-by diesel generator is not available.

As for the ventilation-technological systems of the emergency control point it is not clear from the documentation whether the design includes a solution protected against induced vibration caused by external impacts.

We cannot accept without reservation the design, in which the ventilation systems of the emergency control point are not designed to be protected against induced vibrations caused by external impacts. In our opinion it must be shown within the framework of the implementation plan that no inadmissible conditions occur in the facilities and systems affected by external impacts due to a ventilation failure and it needs to be shown which measures can prevent such conditions. If it is not possible to prove this, the design of the ventilation system of the emergency control point must be designed so that it is protected against induced vibrations caused by external impacts.

2.9 Water Steam Circulation

The steam power system consists of the two equal semi-load branches – each of them is engaged in the power and process steam generation. In principle each branch is assigned to one of the two steam generators of the double-block facility. However, it is possible to change-over to the steam generator in order to maintain continuous operation in the case of failure or revision of the respective branch. For this purpose, the fresh steam side is provided with a cross-connection. This connection is made at the fresh steam side after the shutoff facilities so that the specific module-related requirements are not affected, *e.g.*, the steam generator blocking at the reactor rapid shutoff, and no safety-technological effects to the secondary nuclear steam generating unit occurred. Further on, we evaluate the functions and requirements placed on the module branch.

2.9.1 Fresh Steam System

The fresh steam line runs from the fresh steam containers through the reactor building over the extended reactor structure to the engine room. The reactor building is equipped with a fresh steam safety valve and two shut-off valves, which close one after the other. Both shut-off valves are controlled by the reactor protection system. The heating control valves and the following shutoff valve are designed in parallel with the locking (closing) station.

The facilities for fresh steam shutoff and the safety valve are designed so that they are protected against loading by external effects. Further requirements placed on the fixtures, *e.g.*, output (discharge) cross-section and control timing are given in the failure analysis and control system.

Two medium-self-controlled quick-action valves are designed for fresh steam shut-off, and work on the principle of discharging and their closing time is about 1 second. They are built as Y-type valves, the buildings of which are welded together /U 2.9–1/. A spring-loaded angle valve with additional pneumatic loading is designed as a safety valve, the system of which is protected against a pressure of 208 bar and dimensioned for a performance of 77 kg/second /U 2.9–3/.

The fresh steam system is operated for the distribution of a nuclear generated steam output of 77 kg/second to each steam generator of the turbine set. No safety-technological requirements are placed on the operation of the fresh steam system. The fresh steam safety valve, which is designed as a medium-self-controlled spring-loaded safety valve, fulfills the requirements of TRD 421 /L 51/.

The steam generator shut-off has safety-technological importance in the meaning of the Safety Criteria 1.1 Section 2 of the BMI Safety Criteria. It forms a part of the active measures of the reactor protection systems and it carries out failure and failure control in the facility. The redundant design given by the two fixtures arranged one after the other complies with the total concept of the facility.

Failures of the fresh steam discharge in the form of positive and negative pressure-temperature and flow gradients can be caused by various failures occurring in the components or controls within the entire area of the fresh steam side of the steam power facility.

An inappropriate increase of fresh steam flow caused by a pressure drop can be controlled by emergency openings of the emergency lines of the turbine setting valves, fresh steam bypass valves, in the process steam branch line or by emergency opening of the fresh steam safety valve. An inappropriate drop of fresh steam flow connected with a pressure increase is caused by a turbine quick-connection, by blocking of the bypass valve or by the defective action of one or both fresh steam shut-off valves.

These failures are described and evaluated in Section 5.3.2 and 5.3.3. None of these failures results in the occurrence of inadmissible conditions in the water-steam-circuit or in the primary circuit. By thermo-dynamic feed back in the primary circuit, usually a failure condition on the secondary end is followed by an output reduction in the primary end or reactor quick shut-off and steam generator shut-off, and the facility is switched over to the secured condition while excessive heat is discharged by the surface coolers.

Failure procedures in the case of a broken fresh steam line are described and evaluated in Section 5.4.2. No inadmissible loading of the components occurs in such a case. Breaking of the steam generator hot piping as a result of a broken fresh steam line is not assumed.

The design of the fresh steam systems up to the FD-shut-off valve (inclusive) is protected against loading resulting from external impacts and it can fulfill the Safety Criteria 2.6 "External Impacts". The activation of the FD-shut-off also ensures that the safe condition of the facility is maintained in the case of a facility failure outside the reactor building.

2.9.2 Feedwater System

The feedwater line is equipped with two electrically driven feedwater pumps with controlled speed that supply the steam generator with feedwater – it is fed through the control station and the following shut-off facilities.

The control station is installed in the extended reactor building structure and consists of a maximum load control valve and a parallel partial load control valve. The shut-off facility is installed in the reactor building and consists of two shut-off valves – closing one after the other – and a shock recoil dumper. Both shut-off valves are controlled by the reactor protection system /U 2.9–2/.

The feedwater system is designed so that it is protected against loading by external impacts – from the steam generator up to both shut-off valves (inclusive).

The feedwater system is used to supply the steam generation with a sufficient quantity of feedwater. The feedwater shut-off prior to the steam generation has safety-technological importance. Analogously, in the steam generator shut-off on the fresh steam side, the corresponding control of the shut-off valves must be designed in order to ensure safe general steam shut-off, so that the reactor seal-off fixtures are implemented with sufficient reliability. The redundant design given by the two shut-off valves – closing one after the other – complies with the general concept of the facility.

Failures in the feedwater supply in the form of reduced flow can be caused by control failure or total feedwater failure in the feedwater pumps or incorrect action of one or both fixtures.

A drop in the feedwater flow rate results in an increase of cold gas temperature in the primary circuit. The failure is detected by the reactor protection system starting with the criteria "Flow Rate (from Primary to Secondary End) higher/equal to 1.3" and "Cold Gas Temperature Higher/equal to 280°C", and RESA and steam generator shut-off will be activated. The failure procedure is described and evaluated in Section 5.3.3. A failure is controlled by the facility; no inadmissible loading of the components occurs.

Failure procedures for a broken feeding line are described and evaluated in Section 5.4.2. No inadmissible loading of components occurs. The breaking of a steam generator hot piping as a result of a broken feedwater line is not assumed.

The design of the feedwater system – from the steam generator connection up to the shut-off valves – so that it is protected against loading resulting from external impacts fulfills the Safety Criteria 2.6 "External impacts". The solution of the feedwater shut-off also ensures that the safe condition of the facility is maintained in the case of a facility failure outside the reactor building.

2.9.3 Steam Generator Relief

In the case of a leak from the steam generator hot piping, a water penetration failure occurs due to a pressure drop from the secondary end to the primary end. It is necessary for the failure control to ensure both the release of the steam generator shut-off in order to stop the further supply of feedwater, and the activation of a quick discharge (emptying) of the steam generator in order to limit the further flow of water in the primary circuit.

To fulfill these important safety-technological requirements, two parallel relief lines are provided as branches of the feedwater line between the steam generator connection and the feedwater shut-off valves. Each relief line is equipped with two relief valves – closing one after the other. The design assumes that with a line cross-section of DN 180 and fixture opening time of 3 seconds, the required amount of water can be discharged within a sufficient time period. The discharged water from both lines is fed to a buffer tank and stored in one of the connected containers. Relief valves are controlled by the reactor protection system according to the starting criteria "Humidity in the Primary Circuit Higher/equal to Max." For its active opening function, each fixture is controlled by loading air from the pneumatic piston so that the fixtures are opened by relief. The fixture is closed by a spring after the pressure is balanced between the secondary and primary end.

According to this function of the safety valve on the fresh steam side, the steam generator relief should serve as additional pressure relief for the secondary end /U 2.9–4/. To control the failures "Incorrect Closing of the FD-shut-off valve" and "Non-opening of the Safety Valve" , the steam generation relief fixtures of the line are separately controlled by the signal "FD-Pressure Higher Than $p_{max.\,2}$" – they are controlled and opened by air which is fed under the piston fixtures. If the pressure drops under p=220 bar, the control valves are activated, the lower piston space relieved and the fixtures are closed again.

In our opinion, the design of the steam generator relief with two redundant 100% relief lines is sufficient. The fixture control – *i.e.* opening function by active measures and closing function by passive measures is designed with sufficient reliability. The steam generator relief is sufficient to limit a water penetration failure when the amount of water reaches a weight of 600 kg (see Section 5.4.3).

We agree with the design, where the function of the second-end pressure limiting is taken over by a line of the steam generator-pressure relief system in order to control individual failures in the safety valve.

By the design of the steam generator relief in the reactor building so that it is protected against loading by external impacts, the Safety Criterion 2.6 can be fulfilled.

As for the evaluated function of the reactor protection system in the case of EVA, we must point out the fact that the steam generator relief takes place during or after EVA. As this is not relevant for safety-technological requirements placed on the plant, and additionally the closing function of the relief fixtures is provided by spring force, there are no further requirements for the concept of the steam generator relief.

Loss of integrity of the relief lines or a faulty opening of the relief line are not evaluated separately, as the failure effects are discussed in the analyses of the feedwater line leakage.

2.9.4 Inlet and Outlet Circuit

For supply and discharging the facility and for operational removal of secondary heat, each module is equipped with an inlet and outlet circuit.

The circuit starts from the fresh steam line prior shut-off facility and runs through double-locks to the two throttle valves /U 2.9–5/. After this point the buffer tank and inlet and outlet condensers are connected. The material is fed through the return-feed pump to the feedwater tank. During supply and discharging operations the steam generator is supplied with water by a feedwater pump and control station.

A shut-off valve and a unidirectional restrictor valve compensation valve From the buffer tank or a compensator, the circulation circuit, which is equipped with a pump and a control valve, branches off a. The circulation line is connected to the feedwater line between the shut-off facility on the feedwater side and the steam generator. The inlet and outlet condenser are cooled by an intermediate cooling system and an auxiliary cooling water system. About 20% of the designed steam volume can be condensed on the inlet and outlet condensers.

The inlet and outlet circuit have exclusive operational functions. Only the shut-off valves in the inlet and outlet lines are of safety-technological importance. These are attached to the corresponding double locks in the fresh steam and feedwater sides and controlled by the reactor protection system. Because of this they are included in the requirement of the steam generator shut-off and can fulfill this function satisfactorily.

The unidirectional restrictor valves in the feedwater line and in the circulation line eliminate backlash effects occurring during the operation of one or both circuits.

The effects of a failure in the inlet and outlet circuit, or subjected breaks and leakages are discussed in the evaluation of failures in the fresh steam system and the feedwater system.

2.10 Cold water systems

2.10.1 Overview

According to the Safety Report /U 1/, the following cold water systems are designed for heat removal from the various components and systems:

- The main cooling water system with a hybrid cooling tower,
- The conventional auxiliary cooling water system PC,
- The nuclear auxiliary cooling water system PE,
- The secured auxiliary cooling water system PE,
- The auxiliary cooling water system for the inlet and outlet circuits PC,
- The conventional intermediate cooling system,
- The nuclear intermediate cooling system KAB,
- The secured intermediate cooling system KAA,
- The intermediate cooling system for the inlet and outlet circuits PG.

Of these systems only the

- The secured auxiliary cooling water system and
- The secured intermediate cooling system

fulfill safety/technologically important objectives, according to the Safety Report. They include cooling of the concrete and reactor structures during the designed operation and after a failure, *e.g.*, excessive heat abstraction after the failure of the main heat sink.

Both systems together form a cooling chain, which is constructed as a two-line one. Both partial systems of the cooling chain are constructed as fully separated for the secured auxiliary cooling water system. The partial systems of the secured intermediate cooling system are equipped with a surface cooler – not spatially separated – in the area of the reactor building. The other components in this system are designed as spatially separated.

Both cooling chains are designed so that they are protected against earthquakes. Only the cooling spots of the secured intermediate cooling system and a part of the pipeline grid are designed so that they are protected against an attack by aircraft.

The other cooling water systems fulfill only operational functions. The operational auxiliary cooling water system and the main cooling water system are supplied with cooling water from a shared hybrid-cooling tower. The main cooling water system is constructed as a two-line-system.

The operational auxiliary cooling water systems and the operational intermediate cooling systems are constructed as single-line-systems.

2.10.2 Main Cooling Water System with the Hybrid Cooling Tower

According to the Safety Report, the main cooling water system is constructed as a two-line-system. Each line abstracts heat from the corresponding turbine condenser. Heated water is fed by two in-line pumps to the shared, automatically ventilated hybrid-cooling tower, which releases heat to the atmosphere. The hybrid-cooling tower consists of dry and wet sections.

Depending on the weather conditions, dry operation, wet operation or combined wet and dry operation is set up.

Electric drives of the main cooling water system are supplied with power from their own installation consumption supply. In addition to the main cooling water system, the conventional and nuclear auxiliary water systems as well as the auxiliary cooling water system for the inlet and outlet circuits are supplied with cooling water from the hybrid-cooling tower.

The main cooling water system abstracts heat indirectly from the nuclear steam generating system. The secured auxiliary and secured intermediate cooling system is available for the operational removal of excessive heat and excessive heat after a failure connected with coolant losses. They fulfill the operational functions exclusively. There are no other requirements put to these systems, *e.g.*, redundancy, spatial redundancy-separation, and redundant power supply.

2.10.3 Intermediate cooling systems

All intermediate cooling systems have the following shared indicators according to the Safety Report:

- The lines form a closed system. A buffer tank is provided for each line on the suction side of the corresponding intermediate cooling pumps – this tank is used for pressure control in the system and compensation of volume deviations.
- De-ionate is used as the cooling medium, which is supplied by the de-ionate supply system for filling and leakage replenishment.
- Replenishing leaked medium is carried out automatically depending on the filling level of the respective buffer tank.
- The acid content of the de-ionate is limited because of protection against corrosion.

2.10.3.1 Secured Intermediate Cooling System Including Surface Cooler

Function

The cooling spots supplied by the secured intermediate cooling system KAA remove heat from the non-insulated part of the reactor pressure tanks, the base of the pressure tank unit and the fuel element discharge pipeline by radiation and convection. In this way, the concrete structures are protected against high temperatures.

In addition to its other functions, the surface cooler removes secondary heat after a failure in the main heat sink. In this way the reactor structures, particularly the reactor pressure tank should be protected against high temperatures – in addition to the protection provided by the concrete structures. The surface cooler must perform this function even under external impacts.

The main heat transfer system is designed for cold processes of reactor operation. Alternatively, it is also possible to use the emergency water separator in the Helium cleaning facility. If both these systems are not available, secondary heat is removed only by the surface cooler.

> According to the Safety Criterion 5.2 "Removal of Secondary Heat after a Failure" of the Safety Criteria for High Temperature Reactors /L 7/, a reliable and redundant system must be designed for the removal of secondary heat after a failure, which abstracts secondary heat in the case of events resulting from failure control and from the states and conditions resulting from the failure under consideration so that the specific failure limits for the activity radiation from the fuel elements and of the loading of the safety-technologically important parts of the facility – required for further failure control – are not exceeded.

The secured intermediate cooling system including the surface coolers connected to the reactor cell forms an integral part of the cooling chain from the secondary heat removal, which fulfills the safety-technological requirements given in the Safety Report. Based on the inherent safety properties of the HTR module reactor, the active abstraction of secondary heat is not required for the control of the designed temperature of the fuel elements or for limiting the reactor pressure tank temperatures and the primary cell concrete structures. This system provides protection for at least 15 hours in the case of a reactor failure /U 2.10–1/.

Our tests have confirmed the framework of conditions for the action of active secondary heat removal (see Chapter 5 "Failure Analysis"). Based on this, we consider the specified function of the secured intermediate cooling system as appropriate and sufficient for its safety-technological objective.

Layout, Design, and Function

The secured intermediate cooling system KAA is constructed as a two-line system. The above given indicators for all intermediate cooling systems apply to both lines.

The system removes heat from the following cooling spots:

- Surface cooler,
- Support of the pressure tank unit,
- Structures on the RDB floor (BE-discharge pipe).

These cooling spots are connected to the reactor building. Heat is fed to a secured intermediate cooler in each of the two lines. The cooling medium is re-circulated by a pump to the cooling spots /U 2.10–2/.

The surface cooler is always installed at a distance of about 10 cm from the inner walls of the reactor cell. The distance to the reactor pressure tank is about 1.5 m. The gap between the surface cooler and the concrete wall is insulated so that no damage can occur to the concrete structures because of hot air in the case of unfavorable vibrations.

The surface cooler consists of eight segments, which are connected to one another by screws. They are installed so that no deformation can occur because of heat expansion. The segments consist of vertical pipes, which are welded to each other over bridges. The pipes are alternatively connected in a continuous cycle to the redundancy 1 and redundancy 2 of the secured intermediate cooling system and to the nuclear intermediate cooling system. According to its redundancy, or system relevance, respectively, the pipes in the lower area are connected to one of the three input collectors. The connection between the pipes and collectors is made as a expansion loop in order to compensate for heat expansion in an axial direction. The collectors are connected individually from the reactor cells and they can be shut-off from an accessible area.

The surface cooler of the module has to remove about 850 kW under normal operation and about 400 kW under secondary heat removal operations, in case the main heat sink is not available. This function can be fulfilled by any of the three designed lines.

The supports of the pressure tanks are cooled to prevent inadmissible heating of the concrete structures. The bases are cooled by both redundancies of the secured intermediate cooling system and by the nuclear intermediate cooling system, while sufficient cooling by any of the three lines is ensured. The individual cooling lines are connected to the collectors of the surface coolers according to their redundancy or system relevance.

The U-shaped cooling pipe cools the structure of the fuel element discharge pipeline in the area of the passage through the reactor cell floor. In relation to the general structure these cooling spots correspond to the pressure tank supports.

The secured intermediate cooling pumps, the secured intermediate coolers and the respective buffer tanks are installed in the reactor building structure. The secured intermediate cooling pumps are arranged in separated rooms. The forward and return lines can be shut off inside the reactor building. Fire hose connecting points are provided between the forward closures and the cooling spots as well as between the cooling spots and return line closures.

Electric driving units are usually supplied by their installation consumption supply source. In the case of their installation consumption supply failure, this function can be taken over by the two-line emergency power supply system.

> In compliance with Criterion 5.2 of the Safety Criteria for High Temperature Reactors /L 7/, the secured intermediate cooling system must also fulfill its safety-technological function in case of simultaneous occurrence of individual failures during maintenance procedures. This requirement can be fulfilled in a two-line-system only on condition that the inherent safety-technological properties of the reactor can ensure that in the case of a long-term failure of the secondary heat removing system the required measures are taken during maintenance procedures to prevent the specified limits being exceeded.

The required inherent safety properties are given in the HTR module reactor, since our examinations showed that the control of the fuel element designed temperature is not affected by maintenance of the active secondary heat removing system and the concrete and reactor structures engaged in cooling process can resist for at least 15 hours. Therefore a two-line-system is sufficient if it is ensured that both partial systems cannot fail simultaneously because of the same event. From this aspect we particularly tested potential interactive effects to the partial systems, as spatial separation does not occur in all areas.

Endangering of the parallel running lines of branch 1 and 2 through the other branch in the case of failure under normal operation, *i.e.* at an operational pressure of about 5 bar, can be excluded because of the low energy content of these lines. This also applies to the collectors and cooling pipes of the surface coolers.

In a failure of the intermediate cooling systems (*e.g.*, due to external impacts) the system pressure can reach the activation pressure of the safety valves of 24 bar because of the surface cooler warming up. In the case of a subjected failure of the collectors in the surface cooler it is not possible to exclude a possibility of the redundancy being exceeded and damaging the surface cooler under this system condition.

We examined a very rarely occurring combination of events when the rated pressure grade of the surface cooler, including its collector, of at least 40 bar is designed as considerably higher than the usual system pressure occurring in such a case. We also examined, if the result of a long-term failure of all intermediate cooling lines can be controlled, if this occurs only once during the entire power plant service life.

The scheme of this combination of events is conditioned by keeping the low failure frequency in the collector of the surface cooler under higher system pressure. In our opinion repeated testing is required. As for the testing method we recommend regularly repeated pressure tests on all the lines of the surface cooler under 1.3 times higher pressure than the highest system pressure designed for the safety valve.

In addition, it must be ensured that the two redundancies cannot be damaged simultaneously by fragments of the building structures or components.

The reactor building must be implemented so that no failure or external impacts can damage any concrete structure more than within a single designed redundancy.

It is possible to exclude, for the primary circuit component, that any failure can cause damage beyond a redundancy. No other high-powered systems occur in the reactor cell the failure of which could endanger the lines of the secured intermediate cooling system.

Based on these reasons we do not object to the fact that no spatial separation of the redundancies occurs in the given areas.

For our safety-technological design evaluation, only the heat-technological design of the secured intermediate cooling system and both related lines of the surface cooler in the area of the designed surface cooler is relevant because design requirements, which are agreed upon for the spatial and operation-technological boundary conditions set forth for the withdrawal of heat flow and the to-be-complied-with temperatures of concrete and reactor structures by the equipment of the primary cell in the reactor building have to be coordinated.

If the results of retesting show, for example, that the cooler output is insufficient to remove the required quantity of heat, the surface cooler can be extended within the framework of the presented concept for the facility. For these reasons we evaluated the designed heat-technological function in detail, and we concluded that any of the three designed surface cooler lines can always remove the heat loss output under standard operation of about 400 KJ/second and secondary heat output of about 850 KJ/second, if the given thermal boundary conditions are met.

External Impacts

According to the Safety Report, the secured intermediate cooling systems are designed so that they are fully protected against earthquakes, while the considered intensity of the particular earthquake is site-dependent.

Other natural impacts such as soil slides, storms, flooding and the impacts of biological organisms (*e.g.*, flocks of bird, shells growing in the cooling water lines) are also considered site-specific.

The area of the cooling spots up to the forward and return line closures is designed so that it is protected against aircraft attack and explosion shock wave. It is possible to cool the cooling spots by fire hoses after an aircraft attack or an explosion shock wave, if the area of the secured intermediate cooling systems, which is not protected against these events, is not damaged. Access to these connections is provided even after these external impacts.

According to the Safety Criterion 2.6 "External Impacts" of the Safety Criteria for Power Generating Facilities with a Gas Cooled High Temperature Reactor /L 7/. The system must be designed so that it is protected against external impacts, including, for example, the parts, which are required for secondary heat removal.

The presented design of the secured intermediate coolers protected against earthquake makes it possible to remove the required secondary heat in cooperation with the respectively designed secured auxiliary cold water systems even after an earthquake. An admissible time delay of at least 15 hours until activation of the secondary heat removal systems allows, among others things, a design limited to a two-line-flow within the cooling chain protected against an earthquake, including emergency power supply, as at least one line can be put into operation within this time period.

We consider the designed protection against other natural external impacts as possible in general. It is necessary to site-specifically prove in the implementation plan that the secured auxiliary and secured intermediate cooling systems are protected against these impacts to the extent that must be considered.

Only the cooling spots, the pipeline grid up to forward and return line closures – inclusive – and the connecting option for the external supply of the surface coolers are designed so that they are protected against aircraft attack or an explosion shock wave, *i.e.* active components of the additional cooling lines are no longer available after the occurrence of these external impacts. For this reason ensuring the secondary heat removal is conditioned by the access to the building parts, to which the mobile components should be connected shortly after an aircraft attack or explosion shock wave, as well as by the availability of a fire pump and hoses as well as the connection of these mobile components in a reasonable time. Based on the design of the reactor building and the admissible delay of 15 hours until activation of the secondary heat removing systems, we consider both preconditions are fulfilled.

2.10.3.2 Operational Intermediate Cooling Systems

The operational intermediate cooling systems remove heat from the following cooling spots:

- The conventional intermediate cooling systems from:
  - The cooling machines,
  - The turbines including the generators,
  - The feedwater pumps.
- The nuclear intermediate cooling system KAB from:
  - The cooler and blower of the Helium cleaning facility,
  - The gas removal system,

  - The primary circuit line blower,
  - The third branch of the surface cooler, the additional cooling system of the pressure tank unit, and the cooling system of the equipment of the RDB Floor (BE outlet pipe).

- The intermediate cooling system for the inlet and outlet circuits from:
  - Both condenser coolers of the inlet and outlet circuits in the case of the operation of "the operational inlet and outlet processes", "operational secondary heat removal" and "switching on/off of the modules during performance operation".

The operational cooling systems always have a single-branched structure and take heat from the cooling spots, pass it through the two parallel-arranged coolers and transmit it to the respective auxiliary cold water systems. From this position the de-ionate used as the coolant is driven back by two parallel pumps to the respective cooling spots. The electrical driving system is supplied with power by its own power supply source.

According to the Safety Criterion 5.1 "Removal of Secondary Heat Consistent with the Operation" of the safety criteria, the system is designed as a reliable system for secondary heat removal consistent with the designed operation. It must be implemented so that the specific limit values of the activity release from the fuel elements and of the component loading of the cooling gas system, activation, connecting of reactor coolants and safety connections are met during the entire operation time under the designed operation.

The conventional intermediate cooling system has no other function than those mentioned above. For this reason we consider it as an operational system without any safety-technological importance.

In addition to supply of the other cooling spots, the nuclear intermediate cooling system removes the respective lost heat from the reactor cell via the respective branches of the surface cooler and ensures limiting of the temperature in the concrete and certain reactor structural parts during the reactor operation. With the designed installation consumption power supply, the systems is also ready to take over a function of secondary heat removal if the reactor is switched off and the input and output circuits are not available. A function of temperature limiting applies in this function also to the reactor pressure tank. If the nuclear intermediate cooling systems failed, all the functions related to cooling of the primary cell and reactor structures, can be provided by redundant secured intermediate cooling system, including cooling up of the both respective branches of the surface cooler.

The intermediate cooling system of the input and output circuits is also designed for secondary heat removal during the designed operation after the reactor was switched off. In this function the system cannot be immediately substituted, as the secondary heat removal via surface coolers of the reactor cell assisted by the nuclear intermediate cooling system or secured intermediate cooling system is possible as described above.

The nuclear intermediate cooling system KAB and the intermediate cooling system of the input and output circuits are in general designed for the functions given in the above mentioned requirement on removal of secondary heat in consistence of the designed operation. These functions, which are important from safety-technological aspects can be satisfactorily fulfilled in case of a failure of the designed secured intermediate cooling system KAA under marginal conditions according all safety criteria. According to Criterion 5.1 it is admissible that the system of secondary heat removal under operation is fully or partially identical to the system of secondary heat removal after failure.

2.10.4 Auxiliary Water Cooling Systems

2.10.4.1 Secured Auxiliary Water Cooling System

The secured auxiliary water cooling system PE is constructed as a double-stranded system. Each branch consists of a wet cooling cell, a secured auxiliary water pump and a secured intermediate cooler. The wet cooling cells are two single-branch adjacent structures close to the reactor building. Secured auxiliary water cooling pumps are installed in these structures. The secured intermediate coolers are installed in the reactor building extension structure.

Heat will be brought to the secured intermediate coolers and discharged to atmosphere through the wet cooling cells. The system is supplied with power from the installation consumption power supply source or – if it fails – from the stand-by power supply source. The system is designed as protected against earthquakes.

> The secured auxiliary water cooling system and the secured intermediate cooling system form together a double-stranded cooling chain. Functions, designs and power supply of the secured auxiliary water cooling system comply with those of the secured intermediate cooling systems (see Section 2.10.3.1). In our opinion, the designed structure of the system can fulfill the safety-technological requirements in co-operation with the secured intermediate cooling system.

2.10.4.2 Operational Auxiliary Water Cooling Systems

The operational auxiliary water cooling systems include:

- The conventional auxiliary cooling system PC,
- The nuclear auxiliary water cooling system PE,
- The auxiliary water cooling system for the inlet and outlet circuits PC.

The systems are constructed as single-branched. Each system includes 2 parallel pumps. These pumps take cold water from the cups of the hybrid-cooling tower (see Section 2.10.2). Cooling water is fed via the collecting branch of each system to the intermediate coolers. Each system supplies two parallel intermediate coolers. After the coolers the lines are connected based on their system relevance. Before the re-circulation line in the hybrid-cooling tower, the lines of the operational auxiliary water cooling branches are connected together with the return lines of the main water cooling system and form a shared collecting line.

The following individual intermediate coolers are supplied by the operational auxiliary water cooling systems:

- Conventional intermediate cooler by the conventional auxiliary water cooling systems PC,
- Nuclear intermediate coolers by the nuclear auxiliary water cooling systems PE,

- Intermediate cooler of the inlet and outlet circuits from the auxiliary water cooling system for the inlet and outlet circuits PC.

The systems are supplied with power by the installation consumption power-supply source. The systems are not designed as protected against external impacts.

The operational auxiliary water cooling systems comply with their function, design and power supply of the respective intermediate cooling systems. They do not fulfill any safety-technological function. In our opinion, the designed layout of the systems is suitable for their operational functions.

2.11 Power Plant Auxiliary Facilities

2.11.2 Hoisting Devices

According to the Safety Report, an overhead traveling crane in the welded steel structure is designed for an assembly and revision activities in the reactor building. The crane reaches to the inner area of the building on one side and to a bracket band of the peripheral protective structure on the other side. The crane is designed as protected against the induced vibration caused by external impacts.

The crane is equipped with a crab provided with three hoisting devices, while the main hoisting device serves moving of transport and storage containers, and the auxiliary hoisting devices for moving of the bigger structural containers. All the control elements are made as adjustable power drives. A working speed can be adjusted by a load within a full range of the speed range in any crane. A crane facility is operated by a mobile floor control unit.

The following hoisting devices are designed for assembling and maintenance activities in the individual power plant buildings:

- Crane at the drive-in to the reactor building,
- Crane in the BE-filling station (reactor building),
- Crane for the T/L-tank top cover (reactor building),
- Crane for the catch pot (reactor building),
- Crane for handling of the structural containers (reactor building, BE-discharge)
- Crane in the assembling hole (reactor building, + 15.85 m),
- Crane in the fuel elements – interim storage,
- Crane in the assembling area of the reactor auxiliary facility building,
- Crane at the drive-in to the reactor auxiliary facility building,
- Crane in the workshop for hot processes (reactor auxiliary facility building),
- Crane for the decontamination area (reactor auxiliary facility building),

- Crane over the E-workshop (reactor auxiliary facility building),
- Crane in the engine house,
- Crane over the feedwater pump structure (engine house),
- Crane over the feedwater tank structure (engine house),
- Crane over the assembling hole in the switching station and emergency supply facility.

According to the function given in the Safety Reports the crane in reactor building and its drives should be of the quality category MH1, and all other cranes of the quality category NNK.

In our opinion, reinforcing of the reactor building cranes and their respective drives of the quality category MH1 and the respective scope of applicability of the KTA Rule 3902, Point 4.3 are correct. All design-relevant requirements and quality-assurance measures are given in Rule 3902/3903 /L 67/ to the required extent, and so there are no additional requirements.

We can confirm the applicant's statement that all other hoisting devices as mentioned in the documentation should belong to category NNK and so have sufficient capacity for load transporting. However, it is not clear from the present project stage that a strike of the load to the activity containing facilities cannot cause release of any radioactive substances – this fact would require reinforcing of category MH 2. Reinforcing of the hoisting devices can be designed in the framework of the detailed implementation plan. According to the reinforcement the design-relevant requirements and quality assurance measures will be determined. However, we have no objections to the concept of hoisting devices.

2.11.2 Other Auxiliary Facilities

The term "other auxiliary facilities" should be understood as comprising the following facilities:

- Air pressure facility,
- De-ionate supply system,
- Auxiliary steam system,
- Heating facility,
- Water cooling system.

A brief description and general evaluation of these systems is given below:

Air Pressure Facility

The air pressure facility consists mainly of a compressor facility in the engine house and a pressure air grid. Operational consuming facilities are connected directly. The safety-technological relevant consuming appliances, such as

- Primary circuit shut-off valves,
- Relief fixtures (steam generator), and
- FD-safety valves

are supplied via the pressure air storage tank.

De-Ionate Supply System

A de-mineralization facility is designed for supply of the various water- and steam-circuits with de-ionate. De-ionate is stored in the storage tanks and fed to the consuming facilities via the de-ionate supply system.

Auxiliary Steam System

The auxiliary steam system supplies the various consumer facilities with hot steam:

- During operation,
- In stand-by status,
- During the start-up and shut-down phase.

Auxiliary steam is collected by a collecting steam line of the turbine during operation. During stand-by condition and starting and ending phase the steam is supplied by the designed steam grid.

Heating Facility

The heating facility serves to supply heat to the buildings in the power plant facility and for service water heating.

Cold Water System

The cold water system is divided by the various functional requirements into:

- Conventional water cooling system, and
- Secured water cooling system.

The conventional water cooling system serves to supply the cooling spots in the conventional and nuclear area with cooling water. C

ooling water is generated by cooling machines in the engine house and it is distributed by shared cooling water conduits to all cooling water consuming facilities. The oxygen content is controlled and limited in the de-ionate in order to protect the heat exchangers and piping against corrosion. Waste heat is removed from the cooling machines by the conventional intermediate cooling system. In general, the system consists of:

- Two cooling machines,
- Two circulating pumps, and
- Two buffer tanks including
- The connected pipelines and fixtures.

The air-conditioning facility in the working, computer and electronic rooms is supplied with cooling water from the secured water cooling system during normal operation and in case of emergency power supply. This system is constructed as a double-stranded system. When one branch is in operation, the other is in stand-by status. Switching over is done manually. In general the system consists of:

- Two cooling machine with air condensation,
- Two circulating pumps, and
- Two buffer tanks, including
- Connected pipelines and fixtures.

In case of emergency the cooling machines are supplied with emergency power supply. They are implemented in the switching stations and the emergency power supply buildings. The air-cooled-condensers are implemented on the building roofs.

As for the above-mentioned auxiliary facilities they are operational systems, except the secured water cooling system. Auxiliary power for the safety-technological relevant consuming facilities is not fed directly, but via preceding storage. In this way it is ensured that in case of auxiliary system failure the function of the safety-technological relevant consuming facilities is observed.

According to its important safety-technological function, the air cooling of rooms in which important safety-technological parts are to be kept in operation even in case of failure, the secured water cooling system is designed as a redundant double-stranded system, connected to the emergency power supply, and housed in the switching station and the emergency power supply building.

In summary we conclude that the concept of the above-mentioned auxiliary facilities is appropriate to fulfill the respective functions and it complies with state-of-the-art of technology.

2.12 Power Systems

2.12.1 Design Rules

The power systems in the HTR two-module power plant facilities under designed operation supply the power consuming facilities of the operational and control systems with power and transfer the power generated in the power plant to the high voltage grid.

In case of failure, the power systems must provide the required power for the power consuming facilities, which are needed for failure control of the designed safety facilities.

Power consuming parts of the safety facilities are connected to the emergency power system, which consists of two spatially separated branches consistent with the operational systems. An additional single-branch emergency power supply is designed in the emergency control point of the reactor building.

Electric power for the consuming facilities connected to the emergency power systems is usually supplied via the installation consumption supply of the power plant. The installation consumption supply facility is supplied either from the generator of module 1 or from the high voltage grid via the main grid connector of the module 1 or the reserve (spare) grid connector. In case of installation consumption failure, it is possible to use the two branch-wise assigned spatially separated emergency power diesel stand-by sets for the power supply of safety facilities. These diesel stand-by sets are designed so that each of them can provide the all required power.

The single branches of the electro-technical equipment of the emergency power system are spatially separated and installed in the switching station and emergency power supply building, which is designed as protected against earthquake. The emergency power supply of emergency control point is installed in the reactor building, which is designed as protected against aircraft attack and exposition shock wave. The electro-technical equipment of the emergency power system is designed as protected against induced vibrations caused by external impacts according to the functional requirements of the power consuming facilities supplied by it.

2.12.2 Power supply

The most important safety-technological requirement of the power supply and its configuration is that it must be possible to reliably supply the safety system of HTR module power plant facility with power from the high voltage grid in order to keep the frequency of requisition of emergency power generating facilities as low as possible.

According to the Safety Report /U 1/ the safekeeping system can be supplied by the high voltage grid as follows:

- By the main grid connection from the grid 110 kV (connecting grid), or
- By the reserve (spare) grid connection from the grid 10 kV (plant grid).

During power operation of the power plant the generated power will be brought to the public main by the main grid connection. According to the general switching layout /U 1/ for both modules it consists of one assigned machine transformer and the following power switch.

The respective generator is connected with the generator switch by the main grid connector of module 1 and via a choke to the short circuit limiter, which connects both 10.5 kV installation consumption supply switching facilities. Only the respective generator is connected to the main grid connector of the module 2.

According to the switching layout the main grid connection of both modules is dimensioned so that the rated output of the generator could be discharged via it.

The reserve grid connection of the power plants consists of a line, which connects both installation consumption supply switching facilities to the 10.5 plant grid. The installation consumption supply facilities are supplied with power by this line when the main grid connection and the generator of module 1 are not available. The reserve grid connection is disconnected from the main grid connection by protective-technological elements.

In case of external impact occurrence which may result in long-term failure of the power supply or of the emergency power system, the safety system will be supplied with power with a 400 V cable connection, available in the emergency control point in the reactor building, which is designed as protected against such impacts. If it is not possible to use this cable connection, the system will be supplied with power by a mobile emergency stand-by generator.

Safety equipment of the HTR two-module power plant facility can be supplied with power either by the generator of the module 1, the main grid connection of the module 1, the reserve grid connection or by stand-by power generating facilities of the emergency power system.

In our opinion, this solution fulfills the requirements set forth in the Section 3.1 of KTA Rule 3701.1 /L 53/, which specifies the general requirements for the power supply of the safety system of a reactor facility. We have no objections to the concept of the power supply.

2.12.3 Installation Consumption Supply

According to the Safety Report, the installation consumption supply facilities (EB-facilities) of the HTR two-module power plant facility are designed as double-stranded facility. All power consuming facilities, which are needed during power plant standard operation, will be connected to it. The following three-phase voltage levels are available: 10.5 kV, 690 V and 400 V. Both 0.4 kV switching facilities of the emergency power system are supplied with power via the two 10-kV-EB-switching facilities by the respective branches. The EB-facility is also equipped with a general doubly inducted 0.4 kV supply distributing system (BJA), which serves for supply of the single power consuming appliances (*e.g.*, hoisting devices) that must be available also during reactor standby.

> Regarding the connection of this general distributing system to the double-stranded EB-supply it must be ensured within the framework of the implementation plan that inadmissible twisting of both branches due to intermeshing cannot occur.

The EB-facilities – and also the consuming facilities of the emergency power system – can be supplied with power by the following supply routes (lines):

- From the generator via the generator switch during the module 1 power operation,

- From the main grid connection of the module 1 via the respective machine transformer, if the general switch is opened,
- From the generator of module 1, when the block load is released (*i.e.* 110 kV power switch is opened) and the block is changed for installation consumption supply,
- From 10.5 kV plant grid, when the generators of module 2 and the main grid connected via reserve grid connections are not available simultaneously.

The main grid connection will be automatically changed over to the reserve grid connection.

According to the Safety Report, the rated output of both generators of the HTR Module is 82.5 MW (cos/phi=0.8). The machine transformers are designed for a dummy output of 103 MVA each.

According to the Safety Report, the EB-switching facilities are separated to both plates of the switching station and emergency power supply buildings and they are installed on level 0 m, together with the branch-related standby power supply facilities.

The reliability of the safety system power supply depends on the reliability of the EB-supply under normal operation. It is limited by the reliability of the power supply of the generator of the module 2 and the 110 kV grid.

Due to application of performance-tested or well proven electric operational means – which are designed as easy-to-install and maintenance-free – it is possible to conclude that a reliability of EB-supply is not determined by the operational means employed. The Safety Report does not set any requirements for a quality of the designed operational means; they must comply with state-of-the-art; and the reliable components will be used.

It can be concluded thereby, that it is possible to fulfill KTA Rule 3701.1, Section 3.2 /L 53/, which sets forth the requirements on reliability.

By selection of the appropriate equipment and application of the protection facilities it is possible to prevent to a situation, when the disturbing effects of the EB-supply – which cannot be eliminated – lead to system failures in the emergency power system. There are no concept-decisive requirements. In our opinion, the requirements set forth by Section 3.3 of KTA Rule 3701.1 /L 53/ on connections between EB-facility and emergency power system can be fulfilled.

The EB-supply is constructed in a away that a load release of the power plant in the module 2 was followed by changing of the turbo-set to the EB-power so the subsequent isolated operation is enabled. Only if simultaneous failure of supply via the main grid connection and from the generator of module 2 occurs, will the reserved grid connection be switched-on. In this way the basic requirements related to state-of-the-art of technology set forth for the power supply are fulfilled, as it is specified *e.g.*, in Section 4.1 of KTA-Rule 3701.1 /L 53/.

The branch-wise shared spatial configuration of the EB-facility and the emergency power system in the switching station and emergency power supply building means that the EB-switching facilities could cause secondary damages on the emergency power systems under the designed external impacts. That is why the EB-switching facilities must be configured so that they are stable even in case of an earthquake with associated shock wave resulting from the collapse of tanks with high energy content. This solution does not pose any technical problems and is included in the Safety Report. Therefore, we came to an opinion that these requirements can be fulfilled by the facilities of EB-supply.

There are no concept-related objections to the layout of the installation consumption facility at all.