2.12.4 Emergency Power Supply

Overview

The power consuming facilities, which should be kept in operation even after EB-supply failure, will be connected to the double-stranded emergency power supply. According to the Safety Report, the emergency power supply is designed so that even one branch is sufficient for required emergency power output.

The emergency power supply is constructed as spatially separated along the single branches in the switching station and emergency power supply buildings that are protected against earthquake. According to the Safety Report each branch consists of the following partial systems:

- One 400 V three-phase rotary current/AC switchgear facility, which is supplied with power by the EB-facility during standard operation – if it is not available, it is supplied with power by the standby power diesel generator that is connected to some consuming facilities, in which a voltage-free pause occurs after the EB-supply failure until start of supply from diesel generator,
- One battery-supported 220 V-DC facility that supplies the power consuming facilities without interruption also in the peak phase of the emergency power diesel stand-by set (partially via the 24 V-DC facilities operated by DC converters),

- A secured 400 V rotary power/AC switchgear facility, which is supplied mostly by 220 V DC facility via a current inverter.

In addition to this, an emergency control point is designed, which controls the HTR two-module power plant if the main control fails. It is installed in the reactor building, which is designed as protected against loading by aircraft impact or explosion shock wave.

The power supply of the emergency control point is designed as a single branch and, according to the switching layout /U 1/ and a function description /U 2.12-2/, it consists of the following partial systems:

- One 400 V three-phase rotary current/AC switchgear facility, which is mainly supplied by one of the two diesel-supported stand-by power facilities,
- One 24 V DC-facility, protected against failure by batteries, which is supplied mostly by a 400 V facility via current inverter.

The 400 V facility serves for long-term power supply of the emergency control point at a failure of power supply facilities in the switching station and emergency supply buildings – with possible connection to a mobile emergency power set.

A concept of the emergency supply of the HTR two-module power plant is designed so that in case of failure of the EB-supply and unavailability of both emergency power diesel stand-by sets, the required electric power for the safety-technological important controlling technologies and operation could be provided from batteries of the 220 V-DC facility for a time period of up to 2 hours /U 2.12–2/.

The measuring lines and power consuming facilities supplied by the emergency control points can be also supplied from the 24 V battery of the emergency control point for up to 15 hours /U 2.12–2/.

A probability of even longer grid downtime with simultaneous unavailability of both diesel facilities is considered by the applicant as hypothetical event. For this case he supposes a possibility of external connection to the power supply of the emergency control point within a framework of operation restriction.

Ways of Operation

If the EB-supply fails, the diesel stand-by set will be started either automatically or manually, and at the same time the AUF order will be transmitted to both switching units – connected in a line – forming a connection between the EB-facility and stand-by power facility – according to the criteria "Frequency in the Stand-by Power Plate ≤ min." or "Voltage on the Stand-by Power Plates ≤ min". Manual back switching must be done after voltage is restored.

> The HTR-module fulfills the basic requirements for a connecting structure between the EB-facility and the stand-by power system, as they are set forth in KTA Rule 3701.1 /L 53/. Based on the longer admissible downtime of the diesel emergency power, there are no safety-technological requirements regarding the immediate availability of the diesel stand-by set, and so we consider the operational automatic procedure of the diesel stand-by set start-up and separating of the stand-by power facility from the EB-facility in connection with a possibility of manual action as sufficient.

However, in this case, we assume that within the framework of further detailed planning, the safety load separator of the EB-facility located in the branch pipe connecting to the respective stand-by power systems is replaced with real power switches, because only those can be operated automatically.

Any of the two diesel stand-by sets uses its own auxiliary systems and an independent local control room, from which it can be started and switched-off manually.

By this a requirement regarding the basic layout of the emergency power diesel facilities is factually fulfilled, which is set forth by the KTA Rule 3702.1 /L 54/.

The double-stranded 220 V-DC facilities will be supplied via the current inverter, if the EB-supply or the emergency power diesel stand-by set is available. If these supply facilities are not available, the stand-by batteries for a parallel operation overtake continuous supply of the connected power consuming facilities in each branch without any interruption.

In this way, the basic requirements are fulfilled, as set forth by KTA Rule 3703 /L 56/ regarding the concept of switching of the facilities.

The secured 400 V three-phase power supply of both branches of the emergency power system is supplied mostly from the branch-relevant 220 V-DC facility via a static current inverter. If this supply source is not available, they can be supplied directly from the emergency power facility of the respective branch by switching over to it.

With this structure, the secured three-phase facility can fulfill the basic requirements set forth on a concept of switching as specified for such facilities in KTA Rule 3704 /L 55/.

According to the Safety Report, the single-branch emergency control point in the reactor building is supplied with power by one of the two 400 V diesel stand-by power distribution systems in the switching station and emergency power supply building, if available.

> The Safety Report does not specify, whether these two supply ways are interlocked. It must be ensured within the framework of the detailed implementation plan that no inadmissible intermeshing of the two branches of the stand-by power system occurs at these supply positions.

Power Related Layout Design

In the documentation regarding the layout design of the diesel stand-by power supply /U 2.12–3/, the required output of the stand-by power diesel stand-by is set at 692 kW. The stand-by power diesel set is designed with an output of about 700 kW.

> We have tested the boundary conditions, under which this output is required, and concluded that this output design provides a designed reserve of about 20%. That is why we consider this design of the stand-by power diesel stand-by set as sufficient.

The designed dimensions of the batteries for non-interrupted 220 V-DC supply is also derived from the power balance /U 2.12-3/. One battery with a rated capacity of 2000 Ah is provided for each branch. The power demand of the DC-facility is specified as 842 A and also includes the power consuming facilities, which are supplied by diode-independently by both branches of the DC-facility.

We evaluated the assumptions on which the dimensions of the non-interrupted 220 V-DC supply were determined. Here also, a design reserve of 20% was considered. Appropriate calculations showed a power demand of about 820 A. A short current share of about 95 A is included, so that the designed sustained current is only about 725 A. Under these conditions, the 220 V battery with a rated capacity of 2000 Ah is sufficient for the power supply of the connected power consuming facilities for at least 2 hours.

When the power supply from the switching station and emergency supply building is not available, the power required for a control of facility condition will be supplied from the 24 V battery of the emergency control point in the reactor building. According to the Safety Report and documentation /U 2.12–4/, it should be designed in a way that its voltage will not drop below the required value for 15 hours.

The designed output balance /U 2.12–3/ – as specified for the 24 V battery – provides the required sustained current of 374 A and a battery with total capacity of 4000 Ah is provided.

Our evaluations conclude that in this case a design reserve of 20% was also considered. However, we concluded that, at the current of 374 A, the battery only operates for about 11 hours before it is discharged. This means that the battery is not capable to guarantee sufficient power supply for the supposed time period of 15 hours.

Based on the submitted documentation, we do not consider the insufficient capacity of the 24 V battery of the emergency control point to be a concept-decisive fault, however, we recommend to revise the battery capacity in the implementation plan.

Reliability

The safety facilities of the HTR-Module are connected to the stand-by power facilities. The reliability with which they are supplied with power depends on a reliability of the grid and the stand-by power generating facilities.

The applicant states in his documentation /U 2.12–4/ that a short-term failure (up to 2 hours) and long-term unavailability of the installation consumption supply and emergency power supply (up to 15 hours) can be controlled within the framework of the designed facility dimensions. Furthermore, he explains that the a sufficiently safe facility condition within the framework of residual risk minimization can be sustained through the possibility of an external supply of the 400 V switching station even for failures lasting longer than 15 hours for hypothetical events with a very low probability of occurrence

The technological control of a complete failure of the power supply of all active safety facilities for at least fifteen hours covers the maximum time period of the power supply failure – determined based on long-term experience, without considering possible supply from the emergency power diesel stand-by sets.

We consider this safety property of the HTR-Module as the main design indicator for the rationale of only two diesel stand-by sets for stand-by power supply. According to the single failure-criterion, a double-stranded stand-by power supply should be considered – for the case that one branch is under reparation or fails. Based on experience with downtimes of the public grid, and based on the above-described power supply timing in the HTR module, this requirement is not of safety-technological importance (see Section 5.3.4).

However, we conclude that the estimated very rare occurrence of a grid downtime for longer than 15 hours does not justify the statement that this is only a hypothetical event. That is why a requirement is set forth that a sufficiently reliable emergency power supply must be available within 15 hours at the latest (see also Section 5.4.3). It means for the diesel stand-by sets and other components and facilities that their specification of reliability cannot be based only on immediate availability of emergency power supply, but it is necessary to consider further long-term measures to be taken for removal or control of the failure.

It is stated in the Safety Report that not all requirements of the KTA rules set forth for the emergency power generating facilities have to be observed because of their reduced safety-technological importance in the HTR module. However, we consider fulfilling the important requirements given in the rules as a precondition of sufficient reliability of emergency power generation and distribution – and the long-term failure of emergency power supply due to combined events of "Grid Downtime and Long-term Unavailability of the Emergency Power Generating Facilities" should not be classified as hypothetical event.

The necessary reliability of the stand-by power supply will be achieved when planning and testing of diesel-stand-by power supply aggregate and stand-by distribution equipment will take into account the quality aspects of KTA Rules 3701.1 and 3702.2.

Since an immediate availability of a stand-by power supply production unit is of low significance from a technical safety point of view, the assumed double-stranded execution is considered to be sufficient provided that the above instructions are adhered to.

2.12.5 Cables and Conductors, Cable Conduit

Cable Types

According to the Safety Report, the cables and conductors of the HTR module of the power plant facility will be calculated so that they meet all relevant requirements. This includes adherence to the specified design requirements and protective measures, as well as adjustments with the expected thermal and other environmental conditions, such as application of the FRNC cables in the reactor building and in cable channels between the reactor building as well as switchboards and stand-by power supply building.

The technical documentation concerning the design of fire protection /U 2.12-5/ indicates that in the area of the primary cell with expected high radiation and high temperature mineral insulated cables (NUM cables) should be used, since they show high resistance against these impacts.

The design of cables generally follows the VDE regulations. If all quality requirements are perfectly met then the required reliability of cables under conventional operating conditions will be ensured.

The use of FRNC cables is appropriate for the above areas. Principally this also holds for the use of NUM cables in the primary cell under the conditions of an environment with high temperatures and high radiation. The implementation plan should contain evidence that the requirement profile will be met. It is our opinion that this is not a decisive requirement from a conceptual point of view.

Cable Routing

According to the Safety Report the strands of cables will be spatially separated.

In the switching station and emergency supply station, cable connections within one strand will follow the screening of cable floors and cable shafts. Connections between the screenings for high and low voltage as well as for direct current occur through wall openings. The cable floor under the electronics console is specified for the cable of the reactor protection system and the technical control.

The cable channels for separate cable strands will connect the electronic equipment of the reactor building, machine room, and of the ancillary equipment room to the control equipment and emergency power supply building. These cable channels are strictly separated from the piping.

The cable routing is also directed separately in the reactor building based on redundancies /U 2.12–6/. Documentation /U 2.12–5/ provides that the cable is laid only in single (zero redundant) electronic equipment parts together with redundant cables. The assignment of zero redundant cables to a cable route of a redundancy will be observed for the entire cable route.

Inside the cable routes, the cables are partially conducted on plank beds. A distance of 200 mm is maintained between the power cable on the one hand and the cables of the technical control facilities on the other.

The cables leading to the emergency control point are principally provided as zero redundant versions.

> This spatial separation of cables of different redundancies ensures that an individual event that could cause a failure does not have a negative impact beyond its own redundancy on other cables. The technological redundancy degree of the cables remains unaffected.
>
> Joint location of redundant and zero redundant cables on the joint route are principally admissible, as the assignment of zero redundant cables to one strand along the cable route will not change. On the other hand it is important to make sure during the detailed planning of the cable routes that an internal event within the facility, such as fire, does not at the same time result in failures of measuring points that are conducted to the emergency control point and in one strand to the control room.

The assignment of cables to an emergency control point as zero redundant cables is admissible, as this corresponds to the redundancy degree of the emergency control point. Since the measuring signals processed at the emergency control point are acquired from a measuring converter that is separated from the rest of the redundancies, the redundancy principle and the principle of spatial separation is not violated by this kind of cable routings, as long as the above explanation of the assignment of one strand measuring points to the cable routes is respected.

According to VDE regulations, a distance of at least 200 mm between the plank beds of the cable route must be observed. The presented cable routing takes this into account.

There are no objections to the intended execution of cable routes.

Cable Grommets

In order to provide the gas measuring points and drives inside the pressure container unit with electric power, only those cable grommets should be used that allow for gas-tight power supply. They are described in document /U 2.17–7/. They are steel plates in which pressure-tight and insulated electricity conductors are fused with glass as an insulator. These steel plates are flanged to container spouts and connected on both sides to the conductors of the respective cable.

The arrangement of these routes at the reactor pressure container and the steam generator pressure should always be arranged at the top of the container area for which a relatively lower temperature influence and reduced radiation contamination can be expected.

The indicated construction principle of the cable grommets has been known for a long time and has proven its value in German light-water reactors. In case of the HTR module, other requirements are placed on the dimensioning especially from the point of view of pressure, temperature, and radiation. In spite of different conditions, these grommets are considered to be applicable also in case of HTR modules. During the construction planning, the lifetime of the grommets should be experimentally established during normal operation as well as in case of anticipated failures.

In principle, we can agree with the opinion of Siemens/Interatom, according to whom the KTA Rule for Cable Grommets (KTA Rule 3403 /L 58) should not be applied to the cable grommets of HTR modules. Because in case of the HTR module the grommets are a part of a pressure-conducting enclosure, we consider it necessary that their integrity is subject to high requirements. Therefore we consider it important that the cable grommets are implemented in such a way that repeated tests of their tightness are possible.

The grommets can be arranged without any technical difficulties so that the above requirements are met. In this way not objections to the concept can arise.

2.12.6 Electric Lighting

According to the Safety Report, the electric lighting equipment of the HTR Module power plant facility consists of normal lighting, of stand-by lighting and of safety lighting. The power supply of the normal lighting comes from the normal grid and the stand-by lighting from the stand-by grid.

Distribution of the normal and of the stand-by lighting can be switched from a low voltage switching station to one of the other power supply strand.

The electric safety lighting is supplied with electric power from an uninterrupted power supply (alternating distribution) of the switching station and of the emergency supply building. It provides electric lighting for escape routes and for partial lighting of the control room.

> The concept of the electric lighting equipment with normal, stand-by as well as emergency lighting conforms to the state-of-the-art technology.
>
> Stand-by lighting supplied by emergency power is available in case of failure of the normal electric lighting. If one of both power supply strands breaks down, then the existing switching station of the electric lighting distribution can be used to switch over manually between normal or stand-by lighting supply strand.

If normal and stand-by lighting break down then the emergency lighting allows for safe orientation within the building.

The construction planning should include the percentage distribution between normal and stand-by lighting in individual rooms, the furnishing of rooms and escape routes with safety lighting, as well as assignment of safety lighting to both strands of the spatially separated switching stations of uninterrupted power supply.

The Safety Report states that even after disruption of power supply facilities of switching stations and of the emergency supply building (*e.g.*, as a result of airplane crash or explosion pressure wave) the emergency control point in the reactor building will still be provided with technical lighting facilities. Construction planning should ensure that in case of such an event the safety lighting is available also in other areas of the reactor building so that safe orientation in this building is possible.

2.12.7 Grounding and Lightning Protection Facility

The grounding and lightning protection facility provided by the applicant in the HTR module power plant facility should protect persons and the power plant against inadmissible excess voltage according to the Safety Report.

The grounding equipment consists of an external and internal grounding facility.

The externally positioned wire mesh serves as an external grounding facility, while the grounding conductor mesh inside the building serves as an internal grounding facility. The internal grounding facility should be interconnected to all equipment and instruments that need to be grounded.

The lightning protection facility consists of an external and internal lightning protection facility.

Collecting facilities and leakages, which are connected to the external grounding facility, represent the external lightning protection of the building. The building with central conducting technical facilities should, according to the applicant's Safety Report, be sufficiently insulated and protected against external interference fields. This is achieved by round steel parts in the respective buildings that are welded to each other and connected to the grounding facility and the metal facades of the lightning protection facility.

The internal lightning protection should essentially be implemented on the basis of the following measures:

- The electronic boxes and the sub-distributors will have insulated screening rails. They will be separated from the referential conductor system but be centrally grounded together.

- The cable carrying constructions will be interconnected in a conductive way and connected to the grounding facility.
- Cables with an additional power-carrying screen will be used as conductive technical cables between the buildings in the outside area and the exposed points of the equipment. The screening will be grounded on both ends of the cable.
- If beneficial, the voltage limiting measures will be utilized within the conductive technical power circuit.

The grounding and lightning measures to be taken by the applicant are appropriate for the protection of persons and equipment of the power plant facilities against inadmissible excessive voltage.

The construction plan of the building should include the details of the size of mesh of the external grounding net and of the collecting facilities, the measures of the external lightning protection of the building with the central conductive technical equipment and the internal lightning protection.

The assumed central grounding of the referential conductor system and the screening rails in the area of the electronic boxes and sub-distributors had been code of practice so far. On the basis of new insight /L 115/ from lightning simulation measurements we recommend that after construction of the facilities without grounding it is important to install some additional grounding so that transient loads are reduced.

We do not consider this point to be conceptually decisive. Adjustments to the actual state-of-the-art of technology can be made within the framework of construction planning.

2.12.8 Communication Facilities

Technical communication systems have been provided for the general communication within the power plant facilities, with the outside world, and also in order to alarm the personnel in case of emergency situations. These systems include:

- Alarm equipment,
- Wireless person search facility,
- Central intercom, as well as
- Telephone extensions.

Requirements regarding communication equipment are specified in KTA Rule 3901 /L 59/ and in Criterion 2.9 of the Safety Criteria for Nuclear Power Plants /L 6/.

In our view communication, equipment mentioned in the Safety Report can be interpreted such that all conceptually relevant requirements of the above assessment criteria are met.

2.13 Technical Control Facilities

2.13.1 Control Room and Ancillary Equipment

Control Room

The control room in the switching station and emergency supply facility houses the servicing, information, and reporting facilities, which are necessary for monitoring and control of the power plant equipment. This equipment is in the control room in one main and one ancillary control stand, which takes into account ergonomic considerations. Apart from this the control room also includes technical reporting facilities to alarm the personnel in case of failures, failure instruments, servicing equipment of the communication facilities, as well as information equipment for fire reporting facility and the room monitoring.

The control room area also includes an ancillary control room apart from the above described control room, computer room and control room anterooms. The ancillary control room houses equipment, which is not immediately necessary for manual block control. This equipment includes measuring boxes for tests and registration equipment for the documentation. The computer room houses a process computer with its subsidiary equipment. The control anterooms serve as an office and social area for the shift personnel, as well as an archive. It is possible to control the access in the control room area from one of the anterooms.

The power supply for the operation, display and registration facilities is provided from a secured 220 V-DC switching station. The process computer has also an ancillary power supply from a 400 V emergency source, so that even in case of a failure in the secured power supply the computer gets the necessary power.

Emergency Control Points

The reactor building is protected against all influences from outside. Inside the reactor building is an emergency control point. This control point serves to control the facility in case of failure of the control room. The relevant data are displayed and recorded so that they display the technological safety condition of the facility and, if needed, provide information about the necessary measures to be taken. Apart from that also radiological and meteorological data are displayed in order to record released radioactivity and to control conditions of its propagation / U 2.12–2/.

Small sphere shutdown system serves has a control function from the emergency control point. Shutting down on the basis of the data displayed at the emergency control point is also a manual measure, which can be carried out in case that the control room is not accessible. The necessary measures are then taken on the spot at the relevant adjuster.

The emergency control point including the measuring technology is supplied with power from a single strand emergency power grid, which is built in the reactor building. However, apart from the power supply from both 400 V emergency power switching facilities power supply is also from possible outside.

The emergency control point has its own air conditioning facility, which is supplied by power from the above emergency power grid.

The emergency control point can be entered through a separate entrance from outside even if the switching station and emergency supply building are not accessible.

Local Control and Reporting Stands

Local control stands have been provided for certain systems that do not have any direct connection with the power plant process. These local control points include all control, display and reporting facilities that are necessary for operation of the relevant systems. The Safety Report contains the list of the systems, which are equipped with the local control stands.

Monitoring and Reporting Equipment

Each operation equipment system at the control stands of the control room includes displays and printers for measured values which inform the personnel of the power plant about conditions of the entire facility so that the measured values of the power plant processes can be controlled. Moreover, measured values on monitors of selected sizes are displayed at central points.

The failure reporting facility in the control room indicates and documents failures in the power plant. These reports are classified as the emergency reports of the category I, which indicate failures of the safety system and the emergency reports of the category II, which inform the power plant personnel about the failures in power plant system. These reports can be read in the control room from data display instruments or verbal reporting displays that are systematically arranged in the main or ancillary control stands. The signaling in the control stands are visual as well as acoustic. The printer is used to document reports.

The above-described design of the control room, of the emergency control points and ancillary equipment can be technically implemented. We are of the opinion that the Safety Criteria for Nuclear Power Plants /L 6/ should be met in order to observe the criteria related to 5.1 "Control and Reporting Equipment" and 5.4 "Switch Control Room and Ancillary Control Equipment".

The control room and emergency control point are separated and functionally independent from each other. The control room contains the equipment, which is necessary for the power plant operation. Activation of the small sphere shutdown system is the only possible measure, which can be taken at the emergency control point and is based on the safety concept of the HTR module. According to this concept, the inherent technical safety construction features of the reactor permits giving up automatic control measures and manual interventions in the technical control equipment in order to control the failure after the protective actions of the reactor have been activated at the onset of the failure. In this way, also in case of an external impact which might lead to a partial or full destruction of the switching station and emergency supply facility, the activation of the small sphere shut-down system at the emergency control point is no longer the necessary switching operation for control of the failure, but it is a measure to achieve transition from the hot switched off condition of the reactor to the long-term safe cold condition under the critical state.

Even though the current planning condition does not indicate any need to introduce additional switching off possibilities in the emergency control point, we do not wish to exclude such considerations from the detailed construction planning. Further extension however is not decisive from a conceptual point of view.

2.13.2 Reactor Protection System

Objective

The reactor protection system is a part of the safety system control technology of the reactor facility and its principal objective in case of failures is to automatically activate the Safety Criteria for Nuclear Power plant /L 6/ through protective actions, so that the condition of the facility is kept within the safe limits until the long-term control of the failure is not ensured by manual measures. This objective is reduced in the HTR module reactor for activation of protective measures after the failure has been established. Measures to control the system during the failure process, *e.g.*, control of the secondary heat abstraction or maintenance of filling positions are no longer necessary.

If the reactor protection system is to fulfill this objective it has to establish process values and the related relevant safety variables and activation signals, while the priority is given to activation signals of the reactor protection system over all other control orders. As a result selection of process values related to failures and the relevant activation criteria and activation signals of the reactor protection system are a product of the failure analyses.

Design Principles

Design principles of the reactor protection system are specified in the Safety Criterion for the Nuclear Power Plants /L 1/ and in the KTA Rules 3501 /L 49/. The reactor protection system is constructed in a way that the joint effect of active and passive safety equipment prevents intolerable effects of failures and that the applicable protective actions are not blocked if necessary.

The essential construction principles of the reactor protection system are as follows:

- Preservation of functions so that the reactor remains under control in case of failures caused by *e.g.*, accidental failure, systematic failure, failure during repairs, consequential failure and some combinations of the above,
- Registration of a failure on the basis of various process variables or on the basis of various measurements, measuring devices or similar measures when only one process variable is available,
- Separation of the reactor protection system from other systems;
- Prior activation of signals related to reactor protection,
- Application of components with upgraded quality of instruments and their quality assurance,
- Testing capability and monitoring.

The basic features of the above principles on which construction of the reactor protection system rests are described in Safety Report /U 1/ and in the description of the concept of the reactor protection system /U 2.13-1/.

Activation Criteria and Protective Actions

The following process variables of the reactor protection system should be recorded:

- Neutron flow,
- Hot gas temperature,
- Cold gas temperature,
- Primary circuit humidity,
- Primary circuit pressure,
- Secondary circuit pressure,
- Primary coolant output,
- Feedwater output.

These criteria should be used directly or in connection with calculation switches to establish fixed limits of safety variables, which, if exceeded, should become a base for the following activation criteria:

- Average range of neutron flow ≥ max,
- Period ≤ min,
- Thermally corrected neutron flow ≥ max,
- Negative gliding limiting value of the thermally corrected neutron flow > max,
- Hot gas temperature ≥ max,
- Cold gas temperature ≥ max,
- Output ratio (primary/secondary) ≥ max,
- Output ratio (primary/secondary) ≤ min,
- Negative gliding limiting value of the primary circuit pressure ≥ max,
- Negative gliding limiting value of the fresh steam pressure ≥ max,
- Humidity in primary circuit ≥ max.

These criteria always activate protective actions:

- Downfall of reflector rods,
- Switching off the primary circuit blower,
- Closing of steam generator.

Other protective actions to be taken as a consequence of a failure:

- Closure of the primary circuit,
- Relief of the steam generator.

Closure of the steam generator relief fixtures occurs after equalization of the pressure between the primary and secondary circuit independent of the reactor protection system.

The process variables that are used for creation of activation criteria, serve to recognize

- Reactivity failures,
- Output failures in primary and secondary circuit,
- Failures caused by loss of pressure in primary circuit,
- Leakage in the heating pipes of the steam generator.

On the basis of the planned protective actions, the respective reactor may be safely switched off, the primary circuit closure released, and in case of the leakage in the heating pipes of the steam generator the excessive water steam in the primary circuit will be limited.

In order to record failure, at least two physical diverse process variables are referred to. If only one process variable is available then for the activation level an accumulation of a high value is necessary. High value accumulation for the activation level is expected when recording a leakage in the heating pipes of the steam generator or a breakage of the fresh steam conductor.

In case of a breakage of DE hot pipes the reactor protection system has only the process variable “Humidity in the Primary Circuit” at its disposal. The pressure of the primary coolant which is also used as a diverse process variable to record failure is not suitable for the HTR module facility due to the fact that in case of a subsidiary leakage in the heating pipes of the steam generator an inadmissible amounts of water steam could penetrate in the primary circuit /U 2.13–3/ before the activation criterion could be derived from the coolant pressure. The process of measurement is treated in the Chapter 2.13.4.

Establishment of a high value accumulation in the activation level or other equivalent measure may follow within the framework of a detailed construction planning, but in our view this does not have any decisive significance from the conceptual point of view.

The requirement related to recording of the failure or creation of activation criteria can be met in accordance with the above construction principles subject to the KTA Rule 3501 /L 49/ and Safety Criteria for Nuclear Power Plants /L 6/.

Layout, Arrangement and Testing Possibility

Every module includes its own reactor protection system. Both systems are constructed independently from recording of measured values up to alignment of protection equipment.

Process values are recorded in the reactor protection system, while in the measure converters the power signals are converted and conducted to the equipment for preparation and processing of measured values in the analog part. After that, starting signals are created from the processed measuring signals in the logic part and after logical connection the activating signals are created.

Recording of measured values of process variables, which are needed for creation of safety variables, is done with triple redundancies, with the exemption of neutron flow measurements, in the middle range. For the medium range, the recording of measured values is done twice. All process variables are recorded analogously and converted in the standardized power signals. This conversion is done for the conventional measured values in the measure converters, which are located in the measure converters rooms of the reactor building. In case of the neutron flow measurement this conversion is implemented in a separate measuring console, which is placed in the switching station and in the emergency supply facility.

The power signals in the analog part are supplied by conventional measuring converters, converted in the voltage signals and either switched directly or through the computer switches to the limit signal sensor of the logic part.

If the analog measuring signal is used for more than one limit signal sensor, then a signal multiplication through the irreversible impedance converter will take place in the analog part. Through this impedance converter it is possible to disconnect irreversibly those measuring signals which are needed in other systems, or which should be displayed in the control room (*e.g.*, on the reactor protection table).

The analog part is constructed with triple redundancy. The console groups are spatially separated and located in the switching station and in the emergency supply facility.

In the logic part of the reactor protection system the logical connection of activating signals is done after logical assessment as well as after activation signals have been created. For this purpose, a dynamically working logic system is applied which is different because of its self-reporting and activation-oriented behavior in case of a failure. In this system a definite impulse sequence is subordinated to a "0" signal. Any other impulse sequence or impulse failure is assessed as a logical "1" signal leading to activation. Those impulses, which are specific for the system, are produced in the synchronizer and after amplification they are led to the redundant switch channels.

The redundant switch channels of the logic part are located in the switching station and emergency supply facility in one room together with the console group.

The activating signals from the logic part act on the control level protection equipment in order to activate protective actions. The alignment of this protection equipment is constructed subject to the spacing current principle. A protective impulse from the reactor protection system always leads to the interruption of the energy supply of the relevant protective equipment. The priority of the reactor protection activation signals and the disconnection of operation signals are hereby secured.

The redundant equipment on the control level is spatially separated and located in the switching station and in the emergency supply facility. The adjusters and drives are inside the reactor building.

The characteristic impact of equipment on activation of protective actions will be treated here only at its basic level. The suggested construction is drawn and described in the documentation /U 2.13–1/ and /U 2.13–4/.

Drives, which are available for each reflector rod, are supplied with power from two redundant six-contact systems. Activation from the reactor protection system interrupts power supply and causes reflector rods to fall.

Redundant six-contact systems perform 2v3 assessments of the activation signals assigned to switch channels of the reactor protection system. The planned construction enables to control active as well as passive defects on the activation level. The common mode defect on the control level can be dealt with on the basis of application of diverse protections.

The primary circuit blower and the closure of the primary circuit takes place always on closure equipment, which is arranged in a row. This equipment is controlled from the reactor protection system through three channels in case of primary circuit blower and through two channels in case of the primary circuit closure.

Defective non-reaction of the switch channel in the reactor protection system has no impact on the safe shutdown of the primary circuit. According to the current principal circuit diagram activation the subordinate active defect activation from the reactor protection system initiates protective action.

Control equipment for closing of the secondary circuits and for discharge of the steam generator always has double redundant construction. The control from the reactor protection system is executed through four channels for discharge of the steam generator, one for each discharge strand, and for the closure of the secondary circuit through two channels, one for each shutdown fixture. In case of activation of reactor protection the closing flows of the shut-down and discharge fixtures of the pre-switched magnetic control valves will be interrupted and the fixtures will move into their safety position. In case of both control alternatives one active defect does not lead to activation of the relevant protective action and one passive single defect cannot prevent the relevant protective action, such as safe discharge of the steam generator or safe closure of the secondary circuit.

Every piece of equipment of the reactor protection system has its components and construction measures designed in a way that they are protected against the external impacts such as earthquake, floods, lightning and storm. The reactor building is not constructed so as to be able to withstand civilization impacts, such as airplane crash or explosion pressure wave, but the switching station and emergency supply facilities include actions to protect reactor by shut-down the module and closure of primary circuit as a result of the external impact (*e.g.*, interruption of the power supply for the safety equipment) so that their activation cannot be prevented in case of need /U 2.13–4/.

The spatial separation of the redundant equipment inside the reactor facility and inside the reactor protection system has been designed as a protection against events leading to failures. The spatial separation has been given up if the events leading to failures cannot prevent activation of protective actions, or if they lead to activation of unequivocal safety oriented protective action.

In order to ensure functional independence of the reactor protection system, the boundaries against other systems have been set by special protective measures. Their principles are described in the documentation /U 2.13–2/. As a result, the reactor protection system is protected against connection to systematically alien potentials of below 220 V by electronic devices and against systematically alien potentials of over 220 V by mechanical and construction measures.

The required signal exchange between various redundancy groups within the reactor protection system takes place irreversibly through the systematic disconnection elements. Disconnection of signals in other systems also takes place irreversibly.

The control room has a reactor protection table which serves to inform the operators about the condition of the reactor protection system. This table shows the measured values of the reactor protection measuring points, reports from the reactor protection system and the feedback from the controlled components. There are also servicing elements to carry out recurrent tests of the reactor protection activation signals.

The reactor protection system is mostly constructed to be self-testing and self-controlled. Testing equipment has been provided for those parts, which are not self-tested.

> We are of the opinion that the equipment of the reactor protection system for recording of measured values, for arithmetical switches, for the measured values connections, for recognition of values outside limits, for logical connection and assessment of activation signals, as well as for creation of activation orders can be constructed with sufficient reliability under the current state-of-the-art of technology. These can be constructed in a way that all requirements concerning activation safety, availability, and testing can be implemented.
>
> Sufficient measures have been taken based on redundancies and spatial separation, as well as on the technical construction of switches to protect against events causing failures inside the facility and within the system, as well as against impacts from outside. These measures should ensure that in case of need the required protective actions can be reliably taken and not blocked.
>
> We have paid special attention to the possibility that the switching station and emergency supply facility might be partially damaged by events such as an airplane crash or explosion pressure waves from the point of view of technical switches construction and spatial separation of the redundant components of the shut-down equipment of the primary circuit blower and the fall-down activation equipment of the reflector rods. In our opinion, selected concepts of the multiple activation and of the activation oriented failure behavior in the logic part of the reactor protection system and switching stations demonstrate the required activation safety also in case of partial damage of the switching station and of the emergency supply facility.

Recurring tests of activation signals of the reactor protection can also be made during full operation. Starting the test of the activation signals for the alignment of the switch-off equipment in order to activate the primary circuit closure and to switch off the primary circuit blower lead to activation of the relevant protective action after the shut-down part of a redundant switch channel has been activated. Additional connection for both protective actions is passed up, since the activation of the primary circuit closure has no immediate negative impact on the availability of the facility, while the design presupposes switching the primary circuit blower off during the operation after the availability limits will be taken into account. There are no objections to this as far as the technical safety is concerned.

The relevant requirements have been met as far as construction and arrangement of the reactor protection system is concerned.

Power Supply

The reactor protection system and the control and switch equipment, which are necessary for the arrangement of protective actions, are diode-disconnected and supplied with electric power by the two 220 V direct current feeder of the emergency power facility. The necessary supply voltage for the spatially separated console groups of the reactor protection system is initially transformed by DC/DC converter to specified values /U 2.13–8/.

The supply of triple strand reactor protection system from one double strand power supply system means that the whole reactor protection system may remain without power already in case of a concurrence of one single event-causing failure due to an accidental failure. Because of the activation-oriented failure behavior of the reactor protection system all reactor protection actions may be activated so that they jointly lead to a safe disconnection of the reactor by downfall of the reflector rods and disconnection of the primary circuit blower to the primary circuit shutdown and to discharge of the steam generator.

Requirements of the KTA Rule 3501 /L 49/ concerning sufficient power supply of the reactor protection system will not be formally met in case of concurrence of one single event causing failure with an accidental failure or with a repair. It is our view that these arrangements are adjusted to specification of technical operation objectives of the reactor protection system, as in case of a total stoppage of power supply the facility will be brought into a safe condition and the reactor protection system will not have to meet any other objectives.

Summary

In our view, the concept of the reactor protection system as described in Safety Report /U 1/ and in detail explained in the design description is acceptable at the current state-of-the-art of technology. A detailed test of components and of the construction of the system within the framework of the technical safety concept is not required from our point of view. The construction principle described at the beginning can be met.

2.13.3 Sampling Equipment

2.13.3.1 Neutron Stream Sampling Equipment and the Primary Neutron Source

The main objective of the neutron stream measuring equipment is to record medium neutron stream density in the core of the HTR module and the detection of neutron stream density transients. The measuring equipment must meet this monitoring objective under all operation conditions from the under critical cold up to nominal load, *i.e.* they must cover the measuring range from about decades 10 to 11. It also provides information about the axial and azimuth operation density distribution in the reactor core. The signals of the neutron stream instrumentation are used on the one hand in automatic systems, such as regulation and reactor protection system and on the other hand they always provide information to operators concerning the condition of the core on display and registration units as well as through alignment of reports /U 2.5.5–13/.

The neutron stream measuring equipment is divided into three diverse measuring ranges so that it can control the range of 11 decades:

- Start-up range,
- Middle range,
- Operation range.

One more loading instrument is provided for loading of the first core.

Every measuring range is executed as an independent measuring channel. The neutron stream detectors are located in metallic guide pipes in the wall of the primary cell. They are brought in the guide pipes from above through the accessible connecting consoles and can be pulled up again from the guide pipes.

A total of five guide pipes are available. Approximately in the middle of the core of two guide pipes there are detectors for the starting range. The detectors can be moved inside the guide pipe after they are no longer needed when start-up has finished, and can be placed in a waiting position with a lower neutron stream.

The primary neutron source is located in the middle between these two azimuth positions, but still on the opposite side of the core. It is located in a reflector rod borehole of the side reflector. Cf-252 source with a half-life of 2.5 is used.

Three guide pipes for the medium range and for the operation range are symmetrically divided and adjusted around the core. Three detectors of the medium range are located always in the middle of the core. In the operation range every one of the three channels always consists of four detectors. In one guide pipe there are always two detectors at the level of the upper core half above the middle range detector and always two detectors at the level of the lower core half under the middle range detector.

The loading instruments are a temporary additional instrumentation. During the loading operation they are located on the upper surface of the core hopper and is taken up with the growing core surface.

Following measuring chamber types for individual channels are used as detectors:

- Loading instrumentation: impulse counting pipes,
- Starting instrumentation: impulse counting pipes (BF3, He 3),

- Medium range instrumentation: gamma-compensated ionization chamber (with boron coating),
- Operation range instrumentation: uncompensated ionization chamber (with boron coating).

The measuring channels of the stationary neutron stream measuring equipment are constructed as follows:

The starting range consists of two identically constructed channels. The measuring range covers the lowest five to six decades. One channel consists of a detector, preamplifier, main amplifier, and logarithmic amplifier. On the basis of further differentiation of the logarithmic signal one signal is finally produced, which is in the reverse proportion to the reactor period and well presents the starting procedure. The medium range consists of three channels. Two channels serve for operation purposes; apart from that the signals for reactor protection system are derived from them. These channels are logarithmic and so are the impulse range channels. Also here, the differentiation of the logarithmic signals enables acquisition of a signal, which is in a reverse proportion to the reactor period. In its lower part, the measuring range overlaps the measuring range of the impulse channel by approximately two decades, while in the upper part it is sufficient for up to approximately double nominal performance. The third channel of the medium range is constructed as a linear measuring channel with the range switch. This channel is conducted to the emergency control point for display.

The channels of the operation range are constructed as linear measuring channels. They cover approximately two decades from 1 per cent nominal load up to approximately 125 per cent of the nominal load. On the one hand, the signals of four detectors of one azimuth measuring position are separately further processed, on the other hand the sum of four detectors is used as a signal for a momentary total performance. The signals of the operation range form together with the thermal correction values the initial values for the reactor protection system and present important information about the reactor performance.

The measuring channels themselves monitor their occurring failures through

- Construction groups of the high voltage control,
- Control of the supply voltages,
- Control of the measuring range channels through the comparator.

The channels can be individually and repeatedly tested also during operation by built-in test signaling while taking into account automatic systems such as regulation or reactor protection. During the shut-down period the routine testing of the measuring detectors takes place, such as recording of the features, impulse height spectra, zero effect measurement, insulation resistance and compensation behavior.

The concept of neutron stream measurement equipment for the HTR module corresponds in its essential parts to the principle and construction of the neutron external measuring system, as it is implemented at KWU pressure water reactors. This concept enables application of an extensively tested system in the HTR module.

The measuring system assures flawless monitoring of the reactor core from the loading of the first core until the nominal load is achieved. The overlap of always two decades between the individual measuring ranges is sufficiently large in order to ensure safe take over of the monitoring from one measuring range into the following measuring range and to ensure this take over by appropriate limit values so than no run is possible without a take over.

Specification of these limiting values in the overlap ranges is done within the framework of the nuclear commissioning tests. The measuring range of the middle range is, according to the data from the applicant, sufficient up to 200 per cent of the nominal load /U 1/. We are of the view that extension of this measuring range up to the maximum possible values at transients is desirable.

The concept of the neutron stream measuring equipment based on the system of KWU pressure water reactors can be implemented from the technical equipment point of view without any problems. Suitable components are available also for the measuring system of the input values for the reactor protection system, which however must be of an appropriate quality. We wish to point out that the impulse range must meet the quality requirements of this system even though, according to submitted documentation, it does not send any direct signals for the reactor protection system. Safe transfer of the monitoring objective from the impulse range to the middle range needs to be secured in the quality required by the reactor protection. An appropriate limiting value of the impulse range must prevent the activation if the take over fails to function. Equally it is our view that, in accordance with the IEC 231 Recommendation /L 116/, a minimum limiting value should be provided for the start-up range in the quality required by the reactor protection so that the start is possible only in case of functioning impulse range channels.

Testing of suitability of the intended detector types for the HTR module included comparisons of the expected gamma dose performance for detectors as they appear at the KWU-DWR facilities, as the same detector types are used there. Gamma radiation does not play any essential role in the impulse range and in the middle range.

In the impulse range the selected detector types, BF-3 and He 3 impulse computer tubes permit extensive electronic discrimination of the disruptive gamma impulses. They have proven their value for this purpose. Description in /U 2.5.5–12/ says that the construction of these channels is planned so that the preamplifier can be located in the connection consoles about 13 m from the detector. On the basis of our experience we do not consider this to be an optimum solution from the point of view of occurrence of disruptive signals of the measuring channels. The preamplifiers should be located as closely as possible behind the computer tube within the guide pipe. Suitable and tested preamplifiers for this kind of a solution are available. Since the channels are released after the preset reactor performance is achieved there is no reason to fear that the radiation could have an impact on the preamplifiers.

In order to meet the IEC Recommendation 231 during the first loading under the shutdown condition the Cf 252 neutron source is placed in the reflector rod bore hole opposite of both impulse range channels. Considering the half life of 2.5 years on the one hand and the accumulation of sufficient neutron sources in the burning elements after reactor operation of several months on the other, relinquishing of the activated secondary neutron sources can be compensated. The use of Cf 252 as the primary neutron source has proven its value in DWR facilities already. The IEC recommendation 231 can be met considering selected locations of the primary neutron source and the impulse range channels in connection with the correspondingly sensitive measuring sensors (measuring sensors for the impulse range constructed from the detectors of diverse sensitivity could be taken into account). Detailed calculation proofs have to be supplied within the framework of construction planning of the plant (screening effects) and of the system (sensitivity of the applied measuring sensors) and their observance must be established within the framework of nuclear commissioning tests.

In the medium range, the failing gamma base is discriminated by gamma compensated ionization chambers. This concept corresponds to the proven procedure in the KWU-DWR facilities. Since the ratio of the thermal neutron stream of the gamma dose performance on location of detectors in case of HTR module is comparable with the DWR facility of KWU, the application of uncompensated chambers in the operation range is also possible.

In its documentation the applicant indicates /U 2.5.5–12/ that the neutron stream measuring equipment has been built to withstand safety earthquake. We wish to point out that according to KTA Rule 3503 /L 162/ only the middle range is provided for failure instrumentation, especially also against the EVI failures. For the medium range this requirement can be implemented with tested components. As far as we can say there are no tested components from the failure resistance point of view for the impulse range and the operation range.

Apart from integral operation monitoring the neutron stream measuring equipment also serves for provision of information about the actual operation density distribution. For this purpose, the operation range detectors are distributed around the core in the azimuth and axial direction so that a total of 12 measured values can be gained and separately further processed. In this way the measuring system provides an overall view of the operation density distribution, which is sufficient from the point of view of neutron physical core features (see the Section 2.5.5) and as a consequence the measurement of the in core operation distribution can be compensated. It our view with this measuring system it is especially possible to detect absorber rods which have failed completely and up to a certain degree also partially.

The signals of the operation range are calibrated on grounds of heat balances for the primary circuit and for the secondary circuit within the framework of the commissioning and later also repeatedly for the actual operation of the reactor. Screening effects of the reactor rods and of the small sphere switching elements are continuously taken into account through measuring corrections based on the actual thermal primary circuit data. In this connection we wish to point out the possible impact of various operation conditions of the surface cooler. Comparable corrections have been applied *e.g.*, in the THTR reactors or within the framework of computer switches in DWR in order to correct the impact of primary cooler medium temperature and they have proven their value.

The monitoring concept of the measuring channels and repetitive tests correspond essentially to the procedure of the KWU-DWR facilities. In case of the HTR module, the additional tests of measuring channels in an integrated test simulator during the operation of the reactor are also planned. This is possible by switching residual appropriate measuring signals to the reactor protection system and regulations.

On the basis of an extensive utilization of a proven principle of neutron external measuring system of the KWU-DWR facility we are of the view that the planned stationary neutron stream measuring equipment of the HTR Module meets the requirements and can be implemented without any problems.

2.13.3.2 Fuel Deterioration Sampling Facility

During the operation the fuel elements are continuously taken from the core. These fuel elements are checked from the point of view of their mechanical integrity and their deterioration is measured. On the basis of these measurements the fuel elements that have achieved the target deterioration of 80 GWd/ton SM are sorted out. If the target deterioration has not been achieved, the fuel elements are returned to the core /U 1/.

The fuel deterioration sampling facility also serves to identify the fuel elements, moderator and absorber elements /U 2.13–5/ during planned unloading of the core for its measurement within the framework of commissioning. The ancillary measuring equipment is used to differentiate between the fuel elements and absorber elements during the first loading /U 2.13–9/.

Specification of the fuel element deterioration is based on a gamma spectroscopic assessment of the radiation of the fission product Cs 137 (Ba-137m). The intensity of the gamma radiation with the energy 661.6 keV is measured and compared with the intensity value which corresponds to the target fuel deterioration of 80 GWd/ton SM. On the basis of the results this decisive measurement forms for a computer a relevant control signal, which initiates either the return delivery in the core or the sorting out of the fission element.

Apart from the Nuclide Cs-137, which is used to determine the fuel deterioration, other nuclides also come into being during the reactor operation, which are used as failure nuclides for measurement of the fuel deterioration. According to the data from the applicant these include the nuclides Nb-97 and J-132 with the gamma radiation energy close to the assessed energy of 661.6 keV, as well as fission products that show high gamma energy and due to the Compton effect determine the base of the 661.6 keV line.

These sizes influence the result of measurements and are taken into account by the assessment computer so that at the end only net counts of Cs-137 are available for the determination of fuel deterioration /U 2.13–6/.

Cooled highly soluble Germanium crystal is used as a detector in the measuring system. It is located outside of the biological shield. The ambient radiation is screened by lead. The measuring position for the sphere is located inside the biological screen, approximately 6.5 m from the detector. The useful radiation penetrates through a steel pipe and a collimator through the biological screen and falls directly onto the detector. In this way on the one hand the counts are reduced to the sizes which can be assessed and on the other hand the base is kept low due to scattered radiation /U 2.13–6/.

The method of measurement of the fuel deterioration on the basis of assessment of gamma radiation of the fission products is a suitable process considering the exact state of the knowledge concerning the rules subject to which the fission products are generated and radioactive disintegration takes place. From the point of view of the objective selection of isotope Cs-137 is reasonable on the following grounds:

- The half-life is sufficiently long so that the radioactive disintegration of the isotope during accumulation and during expulsion process can be neglected. The errors during accumulation are under 1 per cent. Apart from this a small but systematic error can be taken into account during evaluation program.
- The isotope in the fuel elements has a high concentration due to a good fission extraction.
- With the emission probability of 80 per cent and the energy of 661.6 keV high detection possibilities is possible for the Gamma-quant.

> Gamma spectroscopic measurements have proven their value and are generally applied as a procedure to determine activity. Sufficiently good detectors with appropriate electronic and standard programs are available for the evaluation of the spectra. High counts, which due to preset precision can be achieved in a ten second measuring time, can also be processed with the appropriate design of electronics and appropriate dead time corrections in the evaluation programs.

The applicant states that the measuring system can record the fuel deterioration of 80 GWd/ton SM with an error of $1\sigma \pm 5$ per cent /U 2.13–6, U 2.13–7/. This error essentially includes the errors caused by bases and neighboring peaks, by the count statistics for Cs-137 peak, the errors in the system calibration, *i.e.* the allocation of measured Cs-137 peaks to a Cs-137 activity in the fuel element, and finally allocation of CS-137 activity to a fuel deterioration value in GWd/ton SM. Calibration and evaluation methods, *e.g.*, dimension of an individual sphere or preparation in the original facility or in the lab and the follow up chemical analyses are conceivable and *e.g.*, in case of an AVR reactor have already been implemented. According to the data about the measuring uncertainty in case of AVR /L 117/, a value with $\pm$ 5 per cent precision can be achieved, however during longer decay time of the fuel elements. The appropriate detailed research of the measurement precision needs to be made within the framework of the design planning.

The above method of the fuel deterioration determination has been applied in the AVR reactor since 1981 /L 117, L 118/. The process has proven itself to be suitable. The experience shows that temporary failures of the measurement and evaluation unit need to be taken into account. Since the measuring system is accessible during reactor operation, the repairs of short duration are possible. The applicant points out that there is principally also a possibility of a "stretch-out" operation for the case of long-term failures.

Also the possibility of incidence of physically irrelevant measuring results of the fuel deterioration measuring facility has been investigated. They may come up as *e.g.*, unnoticed partial failures, electronic drifts, etc. /U 2.5.5–11/. We limit the anticipated possibilities to the impact of these effects within the tolerable indicators through the plausibility monitoring which is continuously done by a computer.

As far as fuel deterioration measuring facility is concerned the decision should also be made concerning the fuel elements, absorber elements or moderator elements and their elimination from the core. If there is no fuel deterioration yet, the distinction can be made between the radioactive and non-radioactive spheres, *i.e.* between the fuel elements and other spheres. This is important during accumulation of the first core. With the preset ancillary equipment it is possible to make this distinction also at short measuring time. Only after activation of the absorber and eventual pollution can the difference between the absorber sphere and moderator sphere be established on the basis of gamma spectroscopic measurements. In case of eventual circulation and new core accumulation due to IBS measurement the distinction between the fuel elements results, absorber elements and moderator elements on the basis of the available fuel deterioration with the fuel deterioration measuring facility can be made. Relevant programs for the evaluation of such measurements can be made and tested whether they meet their objective without any problems.

2.13.3.3 Conventional Sampling Equipment

Objective and Application

The measuring equipment for conventional values measures the operation values of the power plant, then it changes these values to electric signals and advances them to further processing in the reactor protection system, control and regulation equipment, failure instrumentation, as well as to monitoring, display and documentation equipment.

Sampling Principles and Sampling Procedure

- Temperature measurements on the ceramic built-in elements of the reactor pressure container

In support of the construction computations for the temperature distribution in the ceramic built-in elements of the reactor pressure container the instrumentation is provided with NiCr-Ni double cover thermal elements, so that it can be located in the floor reflector, side reflector and in the thermal cover shield. The thermal elements are placed in the grooves or boring holes of the graphite or carbon bricks and fixed with graphite screws or plugs. The cable lead goes from pressure glass leads in the head area of the reactor pressure container through the gas tight leads and thermal cover shield in the opening between the core container and side reflector. The transition from the compensation conductors goes to the external connectors of the pressure glass leads.

The thermal elements are arranged in such a way that information about temperature profiles can be obtained in radial, axial and azimuthal direction.

Of special interest are the measurements of hot gas temperature at the core outlet, as well as in different elevations of the side reflector. The comparison between the temperature distribution in the floor reflector and the temperature at the steam generator inlet allows conclusions about stream bypasses between both measuring points, while the measurements in the side reflector enable detection of the temperature at individual elevations under various facility conditions and operations.

The instrumentation in the ceramic built-in parts is dimensioned as a lost instrumentation. It is not replaced after a failure as it serves only to verify the dimensioning calculations during commissioning and testing phase of the facility, as well as during the first full operation. The double-coated thermal elements always evaluate the measuring signal of one element, while the second element is connected in the accessible area to the cable so that after a failure of the first thermal element the reserve element can be connected. The durability of the instrumentation is estimated to be several years.

The instrumentation is expected to provide information about facility parameters and the facility behavior and the acquired data sets allow monitoring of the integrity of the hot gas flow on the basis of process values of the reactor performance, primary coolant output, reflector rod disconnection, and primary coolant temperature.

– Temperature Measurement at the Metallic Built-in Parts

The thermal load of the metallic built-in parts is monitored by temperature measurements. The jacket thermal elements are distributed on the core container in various altitudes. Further measuring sensors are located on spouts of the sphere off-take pipe in the area above the thermal cover screen and on the outer wall of the reactor pressure container.

While one part of the instrumentation is considered to be the lost instrumentation with the measuring sensors welded to the surface, the other measurements of the container monitoring are provided for the entire lifetime of the facility. The thermal elements can be disassembled for this purpose and replaced during the facility standstill.

– Hot Gas Temperature Measurement

The hot gas temperature is measured in front of the entry in the hot gas stream in the steam generator. The NiCr-Ni-jacket thermal elements in the thimble pipes are provided for this purpose protruding in the hot gas room above the cover plate between the cold gas room of the steam generator pressure container and the hot gas stream. At the same time the gas-tight welded thimble pipe of this construction forms the primary circuit closure and enables replacement of thermal elements while the primary circuit is closed. Double thermal elements are used so that in case of a failure of an individual element the reserve elements can be switched on by clamping connection.

– Cold Gas Temperature Measurement

The measuring sensors for the cold gas temperature measurement are located in the welded immersion sleeve of a flange in the upstream area of the cooling gas between the steam generator and primary circuit blower. The failure of thermal elements can be remedied by clamping on the reserve elements or by replacement of the defective sensor.

The temperature measurement and methods applied in the HTR Module procedure have been in use in a number of nuclear power plants for several years already and have stood the test. The available measuring sensors and measure converters are suitable for the above applications. We are of the opinion that the requirements placed on the components of the temperature measurements can be met.

Extensive instrumentation of the reactor pressure container which is available for commissioning and testing phase of the facility and its built-in parts are in our view capable to bring proof that the construction limits at all conditions and operations of the facility are kept. They also provide sufficient feedback concerning temperature distribution on the reactor pressure container so that even a reduced number of measuring values bring sufficient information about the thermal load.

– Humidity Measurement

Primary coolant humidity is monitored in order to detect the hot pipe leakage inside the steam generator and the related penetration of water or water steam in the primary circuit. Two measuring procedures with various objectives are applied /U 2.7–5/.

Penetration of a small amount of water or water steam is detected by a measuring arrangement of infrared process photometer and a gas analysis. Water that penetrates into the primary circuit produces CO as a reaction product and its concentration is recorded in the primary coolant by an infrared process photometer. The measured gas goes in the optically permeable measuring cell through which a light ray is lead. The absorption of a narrow infrared band in the measured gas is compared through the interference filter with a reference band. The comparison of both signals enables qualitative evaluation of CO concentration in the primary coolant.

In case of growing CO concentration, an additional gas analysis supported by a gas chromatograph has to be made in order to establish if the penetration involves water or air. In case of water penetration, H_2 concentration will be growing.

While the infrared process photometer is continuously operating, the gas analysis is done on a discontinued basis. This process is suitable for detection of small leakage amounts. The measuring range is in the order of 100 to 1000 vpm.

The test gas is extracted from the primary coolant on the pressure side of the primary circuit blower. The test gas flows without any special cooling in the measuring arrangement and after that it goes again in the primary circuit on the intake side of the primary circuit blower.

The ancillary blower supplies the measured gas during the standstill of the blower. The overall measuring arrangements with the appropriate electronics and ancillary blower are located in the measure converter rooms.

The hygrometers are applied in order to detect larger humidity measurements. These hygrometers contain a sensor consisting of an aluminum strip covered with a porous aluminum oxide layer and a steamed gold layer as a counter electrode. This arrangement forms a condenser with the oxide layer as a dielectric. This dielectric behaves in a hygroscope way, *i.e.* depending on the dominant water steam partial pressure more or fewer water molecules penetrate in the oxide layer and change the capacity and resistance of the condenser arrangement. The appropriate control and evaluation electronics supplies the sensor with an alternating current, records the impedance changes and transfers them in the standardized output signal.

Since the oxide layer has only a small thickness (0.3 μm), very soon the balance between the ambient humidity and the water contents of the oxide layer can be set and the sensor reacts within a few seconds. The sensors are provided with dust filters in order to prevent that the graphite dust brought in with the primary coolant does not influence the oxide layer.

The measured gas is taken by the instrumentation spout on the pressure side of the primary circuit blower and after it passes through the test arrangement it is brought back to the primary circuit by the intake side of the lower part of the steam generator pressure container. As a result, the differential pressure on the primary circuit blower is driven through the measurement arrangement. If the blower is at standstill the ancillary pump drives the gas stream in the opposite direction.

Since the working temperature range of the humidity sensor are far below the cool gas temperature, the measured gas is cooled by a measured gas cooler to about 50°C. The measured gas temperature and the measured gas output is regulated and monitored by electronic equipment.

The leakage detection procedure is not of special importance as no diverse measuring procedure is available for detection of water leakage in the primary circuit.

The process of leakage detection with the help of infrared process photometers and a follow-up gas analysis is discontinued and used only for small leakage amounts which cannot start off automatic protective action. On the other hand, the humidity measurement with the help of aluminum oxide sensor meets the starting criterion of the reactor protection "Humidity ≥ max" and switches the reactor quickly off while releasing the pressure off the steam generator. As a result, increased requirements are placed on the technical execution of this measuring equipment from the point of view of redundancy, safety against undetected failure, and instrument quality.

The humidity measurement with the aluminum oxide sensor is a well-tested arrangement, which has been applied in high temperature reactors for some time and has proven its value.

We are of the opinion that the requirements subject to KTA Rule 3501 /L 49/ can be met under the condition of redundancy construction, redundancy separation assurance, timely detection of a failure of measuring arrangements and assurance of equipment quality and especially under the condition of timely replacement of sensors with reduced sensitivity due to aging.

– Pressure Measurements

Pressure and differential pressure measurements are performed with the aid of pressure measuring converters, which are arranged in the measuring converter rooms outside the primary cell and connected to the pressure measuring points through the working pressure conductors. The application of Venturi pipes or standard measuring diaphragm in connection with the differential pressure-measuring converter enables determination of the output in the pipeline.

– Fill Condition Measurements in a Small Sphere Shut-Down System

In order to establish the filling condition in the containers of the small sphere shutdown system, the capacity measuring procedure is applied. The condenser surface is created by the surface of the container middle pipe and the charge of the small spheres, while the dielectric is created by the insulation layer on the middle pipe. The capacity of this arrangement is directly proportional to the filling condition.

– Position measurements

In order to detect position of the reflector rods the movement of the rods is taken over through the gear flanged to the motor drive and conducted to the position display equipment with the rotation indicator. The upper and lower end position of the reflector rods is also monitored by the inductive sensor.

Arrangement, Redundancy, Testing

The measuring value sensors are connected by measuring conductors to the appropriate measuring converter. The measuring converters are arranged depending on the redundancy in separated measuring converter rooms. Connections between the measuring value sensors and measuring converter rooms are executed in the shortest possible route while the redundancy separation principle is observed.

The pressure glass bushings are provided for signal connections of the instrumentation leading from the primary cell to

- Reflector rod drives,
- Stock-piling container for small sphere shut-down elements,
- Ceramic core built-in elements,
- Blower insert unit and the blower flap on the blower pressure container part.

The following values of the steam generator are measured and recorded on various altitudes of the steam generator spouts:

- Steam generator output measurement,
- Humidity measurement in the primary coolant,
- Steam generator pressure measurement,
- Hot gas temperature,
- Cold gas temperature.

The working conductors leading from the primary cell can be closed by two fixtures located one behind the other. All parts containing primary coolant of the measuring equipment located in the measuring converter rooms are enclosed in the boxes which are monitored from the point of view of pressure and temperature increase, so that a leakage in this equipment can be detected.

For the testing of measuring equipment connectors and testing spouts are attached to the appropriate components so that the testing of measuring circuits is possible on the basis of given physical values (*e.g.*, pressure) or simulation of electric signals (*e.g.*, thermal tensions).

Measuring procedure and measuring principles for the HTR two-module power plant are principally appropriate. These processes have mostly been used in other core technical facilities, which have the operationally tested instrument systems available.

We are of the view that the requirements placed on operational measuring circuits and technical safety can be met as far as following features are concerned:

- Technical suitability for measurements,
- Diversity of instruments,
- Redundancy,
- Testability,
- Arrangements subject to ambient conditions on the place of installation.

2.13.4 Control and Regulation Equipment

Objectives

The control and regulation equipment keeps within the set limits or changes subject to a pre-defined program the procedural variables of all operational processes in a power plant (changes in performance, starting and run-out).

Protection against inadmissible operational conditions is provided by bolts/locks, which are a part of the control and regulation equipment (active – protective locks and passive locks), as well as by protective limits.

Regulations of the Nuclear Steam Production System

In a facility consisting of two HTR modules the regulation takes place in blocks with the thermal capacity of individual modules being the preset value. The capacity of individual modules depends on the shares of electric generator capacity and the thermal process steam capacity. The capacity is given so that all other process variables (fresh steam temperature, hot gas temperature) are conducted through the regulation circuits subordinated to the block regulation. The regulation equipment, which is subordinated to the block regulation of the nuclear steam generation system falls into following categories:

- Capacity regulation,
- Hot gas temperature regulation,
- Fresh steam temperature regulation.

The capacity of the HTR Module is regulated by supply water output, with the actual capacity being recorded from the value of the fresh steam pressure, temperature and output in the computer switch. After comparison with the expected value, which is set by the block regulation, the deviation in the regulation causes adjustment of the supply water valve regulation. Regulation of rotations of the supply water pump through the supply water regulation valve will keep the differential pressure constant.

Hot gas temperature regulation is done indirectly by regulation of the neutron stream, which can be set by the reflector rods. Planned values are set by two transferable functional transducers, which depending on the selected operation mode of the facility (full capacity operation or long-term partial capacity operation with reduced hot gas temperature) generate expected value for regulation of the position of reflector rods.

The fresh steam temperature regulation is done on the basis of primary coolant output, which is set by rotations of the primary circuit blower.

Regulations of the Water Steam Circuits

The objective of the regulation of the water steam circuits is to distribute thermal energy supplied from the HTR module to the equipment of the process steam delivery and to power production so that under all operating conditions the thermal energy is utilized to the maximum. In case of reduced thermal capacity of the reactor the generator output will be reduced.

The regulation of the water-steam circuit includes

- Fresh steam pressure regulation,
- Process steam temperature regulation,
- Pressure regulation for the process steam,
- Pressure regulation of the supply water container,
- Counter pressure regulation of the MD turbine,
- Regulation of the turbine rotations.

In order to protect thermal load due to fast transients the preset value equipment will be introduced so as to limit the speed of change of expected values of the turbine rotation and fresh steam pressure.

Locks

Depending on their role active or passive locks is applied in order to protect the facility against defects or from undesirable operation conditions.

The active locks (protective locks) close or switch off the aggregates depending on the incidence of the activation criteria so that the damage can be prevented.

The passive locks are released after conditions for appropriate switching of aggregates have been met.

From the priorities point of view the locks principally overrule manual or automatic orders.

Limitations

Subject to KTA Rule 3501 /L 49/ the objective of the limiting equipment is to limit the process variables at the maximum admissible value.

The HTR power plant includes following limiting equipment:

- Capacity limitation,
- Limitation by rod engagement.

The objective of the capacity limitation is to limit the capacity of the power plant to 100 per cent in the long run.

The limiting value of the capacity limitation is specified to 105 per cent of the thermally corrected neutron stream. The error of the measurement is taken into account by selection of a lower adjustment value. The reaction of the capacity limitation will cause reduction of rotations of the primary circuit blower and following that the capacity reduction.

The primary circuit blower will be switched off and the fast shutdown of the reactor activated if this measure does not lead to an expected capacity reduction within a specified time.

The capacity limiting equipment is executed in two channels.

Limitation by rod engagement is to ensure sufficient shutdown reaction at any time and under all conditions of the facility, which is achieved by a fast shutdown of the reactor brought about by reflector rods.

The maximum admissible immersion depth of the reflector rods is secured by regulation equipment, which is available for each reflector rod. A fixed limiting value sets the minimum distance of each maximum admissible immersion depth. If the immersion depth of one rod is below the limiting value of the minimum distance then the position adjustment that is independent of the rod position regulation activates the emergency warning of the category 1.

From the signal of a deep position of the reflector rod a signal for supply blower of the small sphere shutdown system is derived so that under this operation condition the supply of small absorber spheres is interrupted.

After the fast reactor shutdown the undesirable supply of small absorber spheres is interrupted as a result of the shutdown of the supply blower.

> The objective of the control and regulation equipment is to ensure safe operation of the power plant under all operation conditions, to release operators on the basis of automated operations procedures, to prevent inadmissible or undesirable operation conditions and to issue appropriate warnings and instructions to the operators.
>
> The principles of control and regulation system of a nuclear power plant have been implemented in a number of KWU nuclear power plants, which are comparable to those of the HTR module and have proven their value.

Depending on their objectives, the differences may come up in the control and regulation equipment of nuclear parts, in regulations in the water steam circuit, in active and passive locks as well as in limitations. Inside these facilities, the priority structure is specified so that safe operation of the facility is possible subject to the principle of defense-in-depth-strategy. Compatible electronic systems are available for control and regulation equipment, which include monitoring and failure reporting equipment thus supporting reliability of operations.

We are of the opinion that the control and regulation equipment of the HTR power plant facility is appropriate and effective and can be technically implemented.

The BMI Safety Criterion 5.3 for Nuclear Power Plants /L 6/, "Control and Shut-down Equipment of the Nuclear Reactor" and the required arrangement of control and regulation equipment can be met.

2.13.5 Accident Instrumentation

The objective of the failure instrumentation is to provide following information to the operating personnel of the power plant in case of a failure or in case of an event, which could lead to a higher release of radioactive material:

- The most important data for identification of plant condition,
- Instruction concerning the failure procedure and the effectiveness of the protection measures,
- Documentation of the failure procedure,
- Information, which would enable evaluation of the radiological impact on the environment.

The failure instrumentation is divided into failure display and failure recording equipment. The failure display equipment includes

- Failure overview display,
- Failure detail display and
- Extended range display.

The objective of the failure overview display is to supply in case of a failure the most important data, which are needed to evaluate the facility condition and the failure impact on the environment. For this purpose selected values will be drawn upon from the total instrumentation of the facility. These values consist especially of the reactor protection measuring values, as well as of other measuring values for the evaluation of the radiological impact on the environment. The measuring ranges of the failure overview display are selected so that the values related to the failure are recorded and tendencies and procedures become transparent. The measuring equipment of the failure overview display are so constructed that their functions are not inhibited by the failure itself and its effects.

The measuring circuits of the failure overview display are redundantly executed. For the measuring circuits, which are not redundantly executed the alternative measurements with the same information contents are available.

The measured values of the failure overview display are available in the control room of the power plant and in the emergency control point. These display facilities of the failure overview display are specially marked to ensure their visibility.

Facilities of the failure overview display are supplied from the uninterrupted emergency power supply.

During the failure the detailed failure display supplies the detailed information, which is necessary for monitoring of the systems and of components, which have been applied in order to control failure and minimize damage. The detailed failure display utilizes the operation instrumentation. The requirements placed on the construction of instruments of the failure detail display depend on the requirements of the systems and components, which are subject to monitoring by the detailed failure display.

The extended range display serves to record the measured data, which indicate that the construction limits of the facility have been approached or exceeded by an unforeseeable event.

The failure display and failure recording equipment of the extended range display is located in the control room and at the emergency control point. The power for the extended range display is supplied from the uninterrupted emergency power grid.

According to the safety report of BMI, Criteria 5.2, the devices for the notification of a disruption and expected event progress should supply sufficient information about the condition of the plant to take the appropriate protective measures for the personnel and the plant, as well as indications about the disruption progress and about the effects on the environment. From the available documentation it can be seen that the construction and design of the disruption instrumentation should take place in accordance with the principles established in KTA Regulation 3502 "Disruption Instrumentation" /L 65/, wherein a selection and adaptation of the measurement variables to be determined takes place according to the requirements of the HTR power plant.

In our opinion, all the requirements with respect to scope, design for the prevention of failures-triggering events, suitability for the purpose, and use of reliable and maintenance-poor devices can be fulfilled with the provided disruption instrumentation.

2.14 Ergonomic Configuration

The control room of the HTR 2-module power plant is located with the adjoining control room, computer room, and control room vestibule in the building of the switching station at +7.10 m. It is divided into

- The main control area with the operations and information part for the start-up and shutdown as well as for the process management,
- The secondary control area with displays and control components for the auxiliary and ancillary systems and the recurring tests,
- The communications area,
- The data recording area.

For different systems are provided additional local control rooms.
The emergency control room is arranged in the reactor building (+12.70 m). It serves during a breakdown of the central control room to monitor the plant with the aid of disruption instrumentation. The only possible control function is the triggering of the pellet shutdown system.
The information display in the central control room should unburden the control room personnel via

- A functional arrangement of the central control room,
- A computer-supported data processing,
- A data display via monitors

of unnecessary data, wherein an essential part is also the preferably automatic safeguarding of a task at a location. For the search of a disruption is provided a staggered system of collective and individual messages. As control components are provided keyboards for visual display units, but also elements of the miniaturized control equipment.

Our evaluation of the ergonomic configuration takes place in view of the relevant regulations and guidelines, wherein above all the "safety criteria for nuclear power plants"/L 6/ can be seen as the basis for the present inquiry. The principles, which expressly concern the ergonomic configuration, can be found in Criteria 1.1 and 2.5. In accordance with Criteria 1.1, should be realized, among other things, "ergonomic measures at the work place." Criterion 2.5 requires: "The work stations and work processes in nuclear power plants should be configured taking into consideration the ergonomic point of view, so that they offer the prerequisites for a safety-relevant optimal behaviour of the employees." The concrete requirements are established in KTA Regulation 3904 /L 60/, as well as in the regulations and guidelines which concern the same.

This investigation has the purpose of testing, while applying all the relevant regulations and guidelines, if the presented concept is suitable, in particular for the fulfilment of the requirements of Criterion 2.5, and in this way can be meet the successful damage prevention in accordance with § 7 AtG within the framework of the ergonomic configuration possibilities.

Essential points of view of the evaluation are:

- Sufficient data display of all the relevant event processes,
- The application of systematic analysis and evaluation methods for new parts of the work system,
- A spacious and functional arrangement corresponding to the tasks to be carried out,
- Maintenance of established ergonomic, optical, acoustic, and climatic requirements.

It is in compliance with the state of the planning in the testing of the concept to evaluate the ergonomic configuration of the workstations in the control room, the emergency control room, and the local control rooms. The requirements of the personnel and labor organizations, which are the object of the business ordinances, for example, are evaluated at a later stage of the planning.

The applicant explains that he will comply with the requirements of the BMI Safety Criterion 2.5 /L 6/. He pursues, therefore, the design principle of unburdening humans via automatic devices as well as routine and also safety-relevant functions. The latter was taken over by devices, which cannot be controlled from the central control room.

Basically, according to our opinion, it is necessary that the personnel under specific operating conditions should be able to safeguard the safety functions. For this reason, all the necessary data should be made available in dependence upon the situation. This does not necessarily mean that for this purpose a data compression should be provided so that the data of the processing computer is displayed "only partially in the arrangement required for the corresponding operating condition"/U 1/. The testing of the data performance in the sense of must start before beginning the construction plan, since the concept of the data display is an essential part of, for example, the room planning. We consider, therefore, that it is necessary to configure this concept in such a way that a complete control room instrumentation for all the process-related events and all the analyzed disruptions is available to the control room personnel in accordance with KTA 3501 /L 49/, 3052 /L 45/, and 3904 /L 60/.

The functional assignment of the tasks in the control room corresponds to the current control room technology of nuclear plants. We would like to note, however, that the corresponding utilization of the medium “monitor” in data display goes as far as in previous plants. The “conversion from parallel to serial operation and data technology” goes here according to our opinion so far that in the sense of KTA Regulation 3904 it can be spoken of “new components of the working system.” As a consequence, systematic analysis and evaluation methods should be used already during the planning of the system to test the solution concepts with respect to the measurement of the human performance possibilities and limits.

With respect to the spatial configuration, a rough calculation of the measurements of the control room and the emergency control room from the drawings yields comparable values to those of newer nuclear power plants. However, it can be seen that, aside from other secondary and ancillary control rooms, especially also the archives and the computer room should only be accessible from the control room. Since the last-mentioned rooms are used also by persons, who are not part of the control room personnel, in a layout of the rooms such as this, according to our experience this produces a constant disturbing “through traffic” into the control room. Even though this solution does not contradict the relevant regulations and guidelines, it is evident that here the intentions of the safety criterion 2.5 are contradicted. Therefore, it is recommended that separate accesses be provided for the archives and the computer room when the control room tract is planned in detail.

The coding of the danger messages provides for the colors “yellow” for Class I and “white” for Class II (danger messages of Class S are not planned). According to DIN 19,235 /L 39/, “yellow” has the meaning “caution,” while “white,” instead, is a general information, which is used for the feedback from the equipment, for example, feedback from the control stations. The selected coding is therefore usually, and insofar "white" is also used with other meanings, more prone to errors than the usual coding such as "red" (Class S), "orange"(Class I), and "yellow" (Class II).

Therefore, we do not agree with the provided color-coding for the danger messages. To color-code the danger messages should be selected "red" for Class I and "yellow" for Class II, similar as in the common practice. Alternative coding can be used if they correspond to the provisions of DIN 19,235.

The embodiment of the work place and work means configuration in the documentation presented by the applicant is sufficient and acceptable, in our opinion. With respect to the environmental requirements result, instead, deviations from the valid regulations. For example, the applicant refers to DIN 5,035 /L 38/ for the illumination and to DIN 1,946, Part 2 /L 37/ for the air conditioning. In KTA Regulation 3904 /L 60/, however, are specified requirements that go beyond the latter. The provisions as to the acoustic in the safety report correspond to KTA Regulation 3904, but are not complete. The acoustic pressure level of the air conditioning unit and the reverberation time, for example, are not discussed. For the design of the illumination, air conditioning, and acoustic of the control room, KTA Regulation 3904 mentions the state of the art that should be used according to our opinion.

Based of the embodiments for the emergency control room represented in the safety report, we find no objections, from our point of view, against the configuration of the emergency control room.

2.15 Devices for Fire Protection

In the HRT 2-module power plant must be fulfilled, aside from the requirements of construction regulations with respect to fire protection, which are contained in the federal land construction ordinances of the lands (for example, /L 3/) and the corresponding issued execution ordinances, also the specifications of the nuclear regulatory system of rules, according to which protective measures against fires must be provided to prevent damages caused by the construction and operation of the plant. The protective objectives reached herein are established in the Atomic Law and in the Radiation Protection Regulation and are detailed further in the "BMI Safety Criteria for Nuclear Power Plants" and the "Breakdown Guidelines." They are directed toward the maintenance of the nuclear safety in the case of a fire and require for the fulfilment special fire protection measures for the devices of the safety system or within the rooms, which contain systems and components belonging thereto, and on the components and containers with activity inventory.

The scope of the measures provided therefore is presented in the safety report /U 1/ and the other technical documentation /U 2.12-5 and U 2.15-1/ to U 2/15-3/. The evaluation of the provided fire protection is undertaken in a separate fire protection assessment, which deals with the adequacy of the individual measures and their effectiveness with respect to the achievement of the protection objectives.

As a supplement thereto, it should be tested within the frame of the safety-related concept evaluation, if all the plant components and rooms which should be especially protected from fire from this point of view have been taken into consideration, and if no fire protection measures are provided, which do not agree with the requirements of the plant safety or the radiation protection. The data necessary for this purpose are contained in the technical documentation "Fire Protection Concept" /2.12-5/ and "Construction and Plant-related Fire Protection Measures" /U 2/15-1/.

In the fire protection concept /U 2.12-5/, in the list of the components and devices, which should be especially protected from fire, are first named the components with possible activity inventory. These are accordingly essentially available in the area where the helium purification plant is arranged, at the fuel element handling, at the wastewater processing and storage, as well as in the ventilation-related plants in the reactor ancillary plant buildings.

Objects to be especially protected are also the components and devices, which have the safety and monitoring functions, and which are important control points from the point of view of safety. The functionality of the entire system should be ensured herein via the corresponding placing of redundant component systems.

The spatially separated installation of redundant devices is realized by the safety-related important components of the instrumentation and control and the emergency power supply, the pipelines and components of the intermediate cooling system, as well as in the redundant cable routes. The installation rooms of the components are configured as fire fighting sections. Along the course of the cable routes, the required separation of the redundancies is essentially reached via redundant cable shafts and channels, which are carried out as own fire fighting sections. In the plant engineering-related areas, the requirement for redundancy separation is fulfilled via a spatially separated installation or arrangement. Finally, additional fire protection measures should be established within the framework of a spatial consideration with unavoidable redundancy approximation.

At the control points, which are important from the point of view of safety, the decoupling of the redundant measurements is reached via a separate conversion in the measurement transducers, which belong to the redundancy, which are arranged in an own measurement transducer rooms. Each one of these measurement transducer rooms is configured as its own fire fighting section.

The assignment of the measurements, which are important from the point of view of safety, to the different redundancies, is derived from the following principles:

- Redundancies I, II, and III for the three-strand measurement technology of the reactor protection system,
- Redundancies I and II for the safeguarded intermediate cooling pumps, the triggering of reactor protective actions, and the disruption succession instrumentation, and
- Redundancy 0 for the cables to the emergency control room as well as all devices constructed with only one strand with safety function.

The safety-relevant staggering of the emergency control room results from the discharge function of the nuclear plant control room, which requires an alternating decoupling also during fire events. This requirement is corresponded in that the function range of the emergency control room with all the plant rooms that belong thereto forms its own fire section within the reactor building. To realize the decoupling from the damaging events in the central control room also during a data logging, for the zero-redundant measuring signals to the emergency control room are used their own transducers. The laying of the cables takes place then together with the corresponding one of the three redundancies I, II, or III

To examine the documents, which describe the fire protection, as to the integrity of the listing of the plant rooms and components, which should be especially protected in the case of a fire, we drew on the following documents:

- Shielding and Local Dose Rate
- Nuclear Technical Classification of the Systems and Components of the HTR, Module Nuclear Power Plant /U 2.3-1/, and
- Nuclear Technical Classification of the E and Instrumentation and Control of the HRT Module Power Plant.

We consider the mentioned documents as a suitable testing basis, because they reproduce the provided safety-relevant classification.

When comparing the list of the components with activity inventory, which should be protected in the case of a fire /U 2.12-5/ with the data in the document /U 3.1-1/, we determined that the majority of the plant rooms that are mentioned in the list are those in which the activity-controlling components are installed. The plant rooms with components that are not attached to an activity inventory such as, for example, the storage of flammable waste in room UKA 02043, or the hot workshop in room UKA 04053, instead, are missing. However, in our opinion, a detailed consideration of rooms such as these cannot be left out of the evaluation of the fire protection.

The examination of the components and devices with a safety and monitoring function also lead to the same result. Here also, the data for the nuclear technical classification have more components of the highest quality classes than are taken into consideration in the fire protection documentation. As an example it is referred to herein to the safeguarded secondary cooling water system PE, which has the same classification as the safeguarded intermediate cooling system KAA. In the fire protection concept, however, is listed only the last-mentioned as a device, which should be especially protected.

We consider here also that detailed consideration with respect to the appropriate fire protection measures is necessary.

We do not assign a decisive conceptual meaning to the aforementioned listed incompleteness, since the rooms and components mentioned in the fire protection documents are only exemplary for all the plant components, which should be considered in this connection, and the transferability of the planned protection measures against fires, which are evaluated in the fire protection assessment, are basically assumed. Within the framework of the construction planning it should be individually checked in view of the room lists if special fire protection measures are required in the plant rooms with activity inventory and/or important safety-relevant devices.

We are basically in agreement with the planned procedure during a fire protection treatment, the important safety-relevant control points, including the corresponding cable installation. However, in the laying of the cables to the emergency control room, as well as the cables of the important safety-relevant control points constructive with merely one strand, it is provided to assign to the same the redundancy 0 and to move it to the cable runs of the redundancies I, II, or III. We have nothing against this combined move, insofar as it is ensured that via an internal plant event, such as a fire, for example, control points can fail at the same time, which are lead to the emergency control room and only via one strand to the central control room. More details with respect to this subject can be found in Chap. 2.12.5.

Finally, we determined that, within the framework of the fire protection concept, the protection of the plant elements, the components, and the operation rooms with activity inventory from the effects of a fire and the protection of the important safety-relevant devices and systems from fire-related failures can be realized to a sufficient extent.

The fire protection measures, which do not agree with the plant safety or the requirements of the radiation protection, are not provided with respect to the current state of planning.

3 Radioactive Materials and Radiation Protection Measures

3.1 Radiation, Activation, and Shield

3.1.1 Radiation of the Reactor Core

The radiation, which is generated by nuclear fission during the reactor operation, can be divided as follows:

- The prompt fission neutron radiation,
- The prompt fission gamma radiation,
- The delayed neutron radiation,
- The absorbed gamma radiation,
- The radiation of the fission products, and
- The radiation of the activated nuclides.

After the reactor is switched off only the following types of radiation remain:

- The radiation of the fission products, and
- The radiation of the activated nuclides.

The spatial classification of the radiation sources is different. The prompt nuclear radiation and the predominant part of the radiation of the fission products are generated in the reactor core. The absorbed gamma radiation and the activated nuclides are produced also outside the core by the neutron field. The activated nuclides, and a small part of the fission products, can then be carried off by the cooling agent into other parts of the plant.

The description of the radiation flows on the margin of the active core, which are decisive for the design of the core shield, which are used by the applicant as entry data for the calculation of the shielding, are contained in the safety report.

We have examined if the applicant has taken into account all the sources of radiation fields, which are important for radiation protection, and if the mentioned source strengths can be valid as a realistic prediction with concern to the radiation field to be expected.

The ANISN calculation code is mentioned in the safety report as the calculation method used by the applicant to calculate these variables. The transport code ANISN is an internationally recognized computer program, which is commonly used for this type of calculations. For this reason, we are of the opinion that the values for the radiation flows mentioned in the safety report represent a suitable basis for the calculation of the shielding of the reactor core.

3.1.2 Shielding and Local Dose rate

Shielding of the Reactor Core

To shield of the radiation field emitted by the core is provided a stratified shield arrangement in the radial and axial direction. The inside layers are essentially formed by graphite of the fuel elements, the boride carbon brick and the core container. In the outer layers, which form the form the biological shield, is used concrete.

In radial direction, from the inside toward the outside, the shield consists of the following layers:

- The fuel element graphite,
- The side reflector,
- The boride carbon brick,
- The core container,
- The reactor pressure vessel,
- The surface cooler,
- The reactor cell wall.

The upward axial shield consists, from the inside toward the outside, of the following layers:

- The fuel element graphite,
- The boride carbon brick,
- The cover reflector,
- The thermal cover shield,
- The reactor pressure vessel cover, and
- The concrete lock cover.

We consider that the data contained in the safety report concerning the intended materials, their spatial arrangement, and expected dimensions are satisfactory in order to evaluate the concept of the core shielding. With the presented shielding concept can be obtained a sufficient reduction of the radiation field. The shield is essentially similar in its construction to those used in other nuclear power plants with high temperature reactors.

We carried out our own estimates of the local dose rate after taking into account the measuring results of the local dose rate in the Hamm-Uentrop high temperature reactor. This lead to the result that a total dose rate (gamma and neutron radiation) of between 10 and 100 μSv/h can be expected over the reactor in the reactor corridor, so that this area is accessible. The local dose rates on the reactor cell wall are of the same order of magnitude as in the reactor corridor. Therefore, the shielding of the reactor core can be considered sufficient.

Other Shielding Measures

The shielding measures concerning activity-conductive components and the calculation methods are described in the safety report and in the comment /U 3.1-1/.

The components and equipment with activity inventory to be shielded are, aside from the reactor core, in particular:

- The installations for fuel element handling,
- The helium purification plant,
- The radioactive wastewater treatment plant,
- The storage for radioactive wastes.

The basic material for all construction shields elements is concrete. Other shielding products or shielding materials are steel and lead, for example, for shielding doors and local shields, as well as special concrete on locations where the standard concrete cannot be used due to shortage of space.

The design of the shield relies on shield calculations and operational experiences from nuclear power plants.

The dimensions of the shield in the individual rooms of the control area are adjusted in accordance with the time during which the personnel is expected to remain therein while the reactor is in operation. Accordingly, the radiation will have to be more or less reduced. In order to differentiate better between the expected radiation levels in the individual rooms of the control area, the control area has been organized into room types in accordance with specific local dose rates and the duration of the exposure.

The classification of the rooms takes place in accordance with the local dose rate at a 0.5-meter distance from the strongest radiation source. Therein it must be taken into consideration that the contribution from the neighboring rooms can amount to up to as much as 20 per cent in dependence upon the design of the shield. It also taken into consideration that rooms without their own radiation sources should be shielded against the radiation sources of the neighboring rooms - so that the class limit within the generally accessible area is not exceeded. The applicant states that a differentiated classification of the different rooms of the control area of the local dose rates should be undertaken /U 3.1-1/ in accordance with DIN 25 440 /L 40/. Accordingly, the upper class limits reach from $D \leq 10$ μSv/h to $D \leq 3$ μSv/h. According to the definition of the Radiation Protection Regulation, areas with $D \geqq 3$ μSv/h belong in the prohibited area, which should be enclosed and visibly identified as such. The safety report includes examples of classification of individual rooms.

The following points of view are taken into account for the above shielding measures of the HTR module radiation protection of the operation personnel /U 3.1-1/:

- The separation of the nuclear and conventional parts of the plant,
- The separation of more and less active components inside the nuclear plant parts,
- The separation of the individual highly active parts from the point of view of maintenance and repair,
- The separation between the component, fixture and operation rooms and the corresponding traffic paths with access points,
- The classification into plant and operational rooms and into room classes with the duration of exposure and shielding measures in accordance with the normal operation requirements.

The applicant discloses in the safety report that, during the normal operation, outside of the control area of the dose limiting values for the operational monitoring area and outside the power plant area, the dose limiting values for the outside operation monitoring area will be maintained.

The evaluation criteria for the design of the shield are included in the Radiation Protection Regulation and in the BMI safety criteria 2.3 and 2.4 /L 7/. The most relevant from this point of view is § 54 of the Radiation Protection Regulation according to which the permanent equipment is dimensioned, so that one person does not exceed the yearly body dose of 10^{-2} under normal operational conditions taking into consideration the exposure time.

We have global data /U 3.1-1/ is in our possession with respect to the method used for designing the shield. In accordance thereto, the shielding computer program POIKE determines the gamma flows and the dose rates resulting therefrom on the upward point in accordance with the "build-up method." From our view, the above named computer program contains the usual methods for the shield calculations. The design of the shield took place with the computer programs ANISN and POIKE and by means of the transfer of operation experience of AVR.

This procedure has already proven its value in shield designs of nuclear power plants.

We have tested, by making comparisons with the shielding measures of already operational nuclear power plants, if the presented shielding concept meets the requirements and we have arrived at the conclusion that the planned shields ensure a sufficient protection against direct radiation from components with radioactive inventory.

One part of concrete structures (walls, floors, ceilings), which are to provide shield against the radioactive radiation, has already been sufficiently measured from other safety aspects (for example, design as support walls against external influences) with regard to the radiation protection.

The above classification of the individual rooms of the control area is a suitable aid for the design of the shield because herein must be taken into consideration, not only the radiation source, but also the activities and exposure times necessary in the area of the radiation sources

According to the descriptions in the safety report, the plant is designed so that in the hygiene tract, staircases, connecting passages, and in the operation and control stations, the local dose rates lay under 10 μSv/h and these rooms therefore belong to the lowest local dose rate class.

In accordance with § 54 of the Radiation Protection Regulation /L 2/, a maximum local dose rate of 5 μSv/h results for long-term exposure locations with an exposure of 40 hours per week.

For the lowest local dose capacity class A, the manufacturer reports an upper limit of 10 μSv/h. It should be taken into account that this value is valid for the most unfavorable room point and the local dose rates lie are under 5 μSv/h in the areas which can be discarded for long-term exposure in comparable plants. The requirement according to § 54 of the Radiation Protection Regulation can be met also in areas where permanent exposure is planned. The detailed testing of compliance with § 54 of the Radiation Protection Regulation does not take place within the framework of the concept evaluation and must take place within the construction planning of the HTR module nuclear power plant.

The presented shield concept, together with the measures described in the following chapter, represents a design that corresponds to the state of the art for the protection of the operation personnel from radioactive radiation. There is sufficient space available in the area of the components and the ancillary and subsidiary facilities for eventually required additions to the shields, which might become necessary in individual cases based on the measurements. We do not envisage any problems here from the point of view of the structural analysis.

A detailed specification of individual shielding measures (especially of the disturbance points, such as ducts, etc.) should take place within the framework of construction planning.

Our control computations carried out within the framework of the concept evaluation have shown that the statements of the applicant about the operational and non-operational monitoring area are correct. In our view, the establishment of monitoring areas outside the operational area can be avoided by the suitable selection of the power plant limits (determination of the distance to the fence).

In summary, we have reached the following results during the evaluation of the shield:

The HTR module does not differ from other nuclear power plants with reference to the principle of the design of the shield. The design parameters of the inventory of radioactive substances on which the shield is based in the systems

- The reactor core,
- The reactor cooling system,
- The series-connected systems, and
- The cooling water systems.

result from calculations with recognized computation procedures and from measurements made in nuclear power plants, which are already in operation. The design of the shields based on these values is sufficient from the point of view of the radiation protection of the personnel and of the environment.

3.2 Radiological Occupational Safety

3.2.1 Principles

The following criteria were applied for our evaluation of the radiological occupational safety:

- Radiation Protection Regulation /L 2/,
- Safety Criteria for Nuclear Power Plants, Criterion 2.4: radiation exposure inside the plant /L 6/,
- RSK Guidelines for pressure water reactors /L 10/,
- General instructions for radiation protection of the personnel (Planning Precaution Instructions) /L 15/.

The compliance with these radiation protection requirements should already be taken into account in the planning phase, so that in the correct operation and in the case of disturbances, it is achieved for the employees that

- Any unnecessary exposure to radiation and contamination is avoided,
- Any radiation exposure or contamination is held as low a possible, even below the limit threshold of the Radiation Protection Regulation.

These radiation protection principles for persons, which are exposed to radiation because of their profession, are valid for both the employees of the power plant itself as well as for outside persons.

In accordance with Criterion 2.4 of the BMI Safety Criteria, the systems and plant parts must be constructed and arranged so that the radiation exposure of the employees during the operation is as low as possible in all necessary activities and maintenance work. As a consequence of this result several requirements with respect to the Planning Precautions Regulation, which relate to arrangements concerning the rooms in the plant, the systems, as well as the components.

For the case of the shut-down and removal of the plant, it should be ensured in accordance with the RSK Guideline on radiological occupational safety already during the planning stage of the plant that the radiation protection principles are taken into consideration also for the case of an eventual removal of the plant parts. Chapter 6 of this expert opinion also deals with the issue of shut-down and removal in particular.

3.2.2 Range Concept

According to the safety report, the dose limiting values for the power plant area for the monitoring outside the plant and on the land of the power plant outside the control area should be met.

The reactor building and predominant part of the reactor ancillary buildings belong to the control area.

The intention of the applicant is to classify the control area rooms according to DIN 25 440 /L 40/. Each class designates one local dose range area whose upper limit increases from one class to the next by a factor of 10. Under this condition, the building arrangement should enable the access to even strongly radiating plant parts at low radiation load at the entrances and exits. This should create preconditions for accessibility of all rooms of the reactor building with the exception of the primary cells during the operation.

An allocation of the rooms to the local dose rate classes has been made on an exemplary basis for some room groups.

As inaccessible areas are discarded the accesses and the rooms of the steam generator and reactor pressure vessel.

We have no objections to the presented concept of areas.

The plant is so planned that that the local dose rate is under 1 x E-5 Sv/h in the connecting corridors and staircases, common rooms, and local control points so that these areas belong to the lowest dose rate class. This value is for the most common room areas sufficiently low. We wish to point out that, according to the Radiation Protection Regulation which concerns the common rooms, the local dose rate must not exceed 5 x E-6 Sv/h. Basically, it must be taken into consideration that this value must be reduced in all common areas when, aside from the external radiation exposure, it can also come to a radiation exposure by the incorporation of radioactive materials.

We have no objections to the planning concept for the classification of the control area rooms since in the allocation of the individual rooms to local dose rates, which takes place during the planning of the construction, if necessary additional shielding measures can be implemented within the framework of the presented concept.

The access to the control area for the reactor ancillary plant building and to the reactor building is provided on the plane of + 7 m in the reactor ancillary plant building. The control area can be also left through the emergency exits.

According to the requirements of the Radiation Protection Regulation, persons remaining in the control area must have measuring devices available, which allow the determination of the body dose and measure the personal dose. Persons staying in the areas where open radioactive materials are handled have to be tested before they leave the control area, to determine if their skin or clothing is contaminated.

According to the data of the safety report, all persons entering the control area are provided with a directly readable pen-like dosimeter and with a film dosimeter. Upon leaving the control area, it is provided that all persons will be tested for contamination with the aid of personal monitors.

At the entrance of the control area are planned a hygiene tract with dressing rooms for the company personnel, the outside personnel, and the guests with "hot" and "cold" showering and washing rooms, as well as locker rooms, toilets, and a room for first aid measures are planned.

> The spatial location of the control area entrance and the arrangement of the neighboring rooms is adjusted are adapted to the requirements. The access to the control area is separated from the exit and can be controlled. The access to the dressing rooms is also appropriately arranged. The locker rooms, contamination controls, washing facilities, and decontamination facilities are arranged according to the appropriate sequence at the exit of the control area.
>
> We are of the opinion that at the entrance in the control area, instead of the pen-like dosimeters, a state of the art electronically assessable and directly readable dosimeter should be made available.

There are no objections with respect to the concept of the control area access with hygiene tract and the first aid room.

3.2.3 Measures Concerning Radiological Occupational safety

According to the safety report and the additional documentation /U 3.1-1/, the applicant intends to take the following measures in order to meet the requirements of the radiological safety at work:

- The provision of the control area with a controlled access,
- Arrangement of the rooms so that the access to the components is not through those rooms which are expected to have a higher local dose rate than that in the target area,
- Provision of radiation protection retention areas and radiation protection doors,
- Arrangement of multi strand nuclear systems in separate rooms (for example, helium purifying plant),
- Application of rapidly removable insulation,
- Planning of shields so that the repairs in rooms with higher radiation level can be prepared and carried out from places with a lower radiation level,
- Planning of a free zone for work preparation with the possibility of application of complete protection clothing and breathing protection devices,
- The partition of the ventilation systems into partial strands so that directed airflow is maintained from the accessible operation areas into the inaccessible plant areas.

In the following part we shall express our view on some measures, which are especially important from the conceptual point of view. The complete and detailed fulfillment of requirements of the Radiation Protection Prevention Guideline /L-15/ will be referred to only within the framework of the construction planning.

A substantial contribution to the reduction of the radiation exposure of the personnel will be achieved via the arrangement of rooms and components. We wish to refer in our evaluation to a few examples:

- Both modules and their active components are separately arranged in primary cells,
- The pumps of the safeguarded intermediate cooling system are spatially separated and shielded in the reactor building annex,
- The concentrate containers for active wastes are individually arranged in shielded rooms of the ancillary building of the plant.

Aids to reduce the work effort on the primary circuit components and the connected systems include

- The preparation for the inspection of the reactor pressure vessel via a camera system as well as a mechanized ultra sound test,
- The use of manually rapidly removable insulation,
- Screw driving equipment for the reactor pressure vessel cover and the building cover,
- The use of a mobile shielding stage above the thermal cover-shield.

These measures are suitable to minimize the necessary time and therefore the radiation exposure of the personnel.

To constructive measures, which are suitable to reduce the radiation exposure of the personnel belong, for example:

- The construction of the components, which reduces the frequency of the repair work and the expanse connected therewith,
- The construction of the containers, which makes possible a good decontamination and complete evacuation.

The concept of mutual shielding of activity-conductive components is for the most part maintained in the control area. An exception is the two wastewater collection containers, which are located in one room. We consider it to be appropriate that during construction planning they should be mounted separately.

The applicant has stated in its safety report and in the additional documentation /U 3.2-1/ data about the way it intends to comply with the Planning Prevention Guideline concerning some essential maintenance work and recurring tests. This applies to the following typical recurring inspections and maintenance work:

- The pressure vessel test,
- Seal tests on the primary circuit,
- The reconditioning of the primary circuit blower,
- Tests on the steam generator,
- The leakage on the steam generator pipe,
- The replacement of filters in the air filters.

A minimization of the dose values will be achieved by the applicant via

- A targeted shielding of the core ("primary cells"),
- The construction of the components to reduce the necessary reconditioning effort,
- The use of manipulator technology,
- The use of special tools (for example, screw driving equipment),
- The use of mobile shields,
- The use of rapidly mountable insulation.

The principles of the Radiation Protection Prevention for recurring tests, inspections, and repair work especially on the reactor pressure vessel are complied with by provided measures.

For the detailed planning of the working processes, which must take place within the framework of construction planning, is also required a break-down of the collective doses according to the individual dose contributions and a detailed description of the individual working steps and local dose rates.

The safety report contains the data and analyses of the disturbances. After certain disturbances, repair work has to be carried out on the systems and components in the plant.

The construction planning should include inspections of the work scope, accessibility of plant rooms, and accessibility of relevant components. During these works in the plant rooms and on the components, the expected local dose rates and the activity concentrations in the room atmosphere should be determined. Also the measures for the radiological occupational safety should be planned and the expected personal doses should be estimated.

The data of the applicant with respect to the measures to be taken in case those large components of the HTR module have to be disassembled are limited to a conceptual overview. Accordingly, all of the unanticipated damages to the main components of the primary circuit, such as the core container, the ceramic core components, the steam generator, and the hot gas pipes can be repaired in reasonable time after the core is evacuated.

In order to evaluate the measures concerning the radiological occupational safety, these general data about the disassembly of the large components are not sufficient. For this purpose would be necessary detailed descriptions of anticipated measures in accordance with guideline /L-15/, a description of the work processes, and of preventive measures for radiation safety as well as an estimate of the work needed and the anticipated collective doses. In our view, these examinations should be made already during the building construction planning stages, so that eventual changes in details of the spatial implementation of the construction can be taken into account.

3.3 Radioactive Materials

3.3.1 Radioactive Materials in the Primary Circuit

Emissions from the Fuel Elements

The applicant has presented in its safety report /U 1/ and in the documentation /U 3.3-1/

- Which mechanisms are to be examined in the activity emissions from the fuel elements in the primary cooling agent,
- With which models is determined the activity emission,
- Which activity emission in the cooling agent is to be expected.

Accordingly, the emission mechanisms can be classified into

- Activity emissions from intact particles,
- Activity emission from the particle decay conditioned by production,
- Activity emission from the particle decay conditioned by the radiation,
- Activity emission from the contamination of the matrix graphite.

When the particles are intact, the coatings of the particle form an effective barrier against the activity emission under normal operation conditions so that no relevant emission rate stakes place on the fuel element surface.

When producing the fuel elements, it cannot be excluded that occasional defects on the particle coating will occur. The applicant indicates an expected value of defective fraction of 3×10^{-5}. The design should be based on a conservative view of this issue and the defect fraction should be doubled to a value of 6×10^{-5}.

In case of defective layers, the essential barriers against the retention of activities are not effective. The transport of fission and activation products takes place then via diffusion from the place of origin inside the particle of the fuel core to the kernel boundaries, then via kernel boundary diffusion directly in the matrix graphite and from there on the fuel element surface. In case of gaseous fission products there is a delay of the emission based on the slow diffusion speed in the fuel kernel. The diffusion along the kernel boundaries and in the matrix graphite takes place relatively quickly. The diffusion speed of the solid fission and activation products is relatively small, both in the fuel core as well in the matrix graphite, which leads to a delay of the emission, and therefore to a reduction of the emitted fraction above all of the short-lived isotopes.

When using the fuel elements inside the reactor can basically be expected an incidence of additional particles defects, for example, due to fuel combustion, fluency of fast neutrons, or the temperature. The applicant states an expected radiation-related particle fraction of 2×10^{-5}. This value is derived as a 95 percent confidence value from irradiation tests with approximately 200,000 particles, which did not show any defects. The further planning will be conservatively based on a fraction of 2×10^{-4}, which is higher by a factor of 10. The transport of fission and activation products from the fuel core to the fuel element surface takes place as in the case of particle fragmentation conditioned by production.

The natural graphite contained in the matrix graphite is polluted with uranium. This uranium contamination caused by the natural composition has as a consequence that a small part of the uranium fission takes place in the matrix graphite, that is, outside of the coated particles. The fission and activation products produced in this way can reach the fuel element surface analogously as in the above mechanism via the diffusion in the matrix graphite.

The applicant puts together the "free uranium inventory" resulting from the matrix contamination and the particles defects caused by production both from the point of view of BE maximum admissible quantities as well as from the calculation of the emissions rate. The interpretation value of 6×10^{-5} which has been indicated for the particle defect fraction conditioned by production also includes the uranium contamination of the graphite matrix. Since the emissions from the defect particles conditioned by production and from the graphite matrix have the same emission mechanisms, both source energy levels are included in the term "free uranium" and the resulting emission rate is calculated on the fuel elements surface.

Apart from the uranium contamination the matrix graphite, the latter is also contaminated with Ag 109. Under neutron radiation, the isotope Ag 110m is produced via the n, γ reaction from Ag 109. The Ag 110m is emitted analogously to the fission product emission from the uranium contamination of BE to an extent that cannot be neglected.

The applicant states in the documentation /U 1, U 3.3-1/ in accordance with the classification according to nuclide groups the emission rates of fuel elements in the reactor core for a selection of the important isotopes. The applicant substantiates the restriction to the stated nuclides by the essential, that is, the radiological relevant contribution to the activity release into the environment. Those fission and activation products, which have not been presented, do not contribute greatly to the radiation exposure due to their low activity rate, their short half-life, the retention in the fuel particle or in the graphite, the retention in the series-connected parts of the plant, or their low biological effectiveness.

The emission rates have been detected with the STADIF and SLIPPER programs. The models used in these programs are described in detail in the documentation /L 122, L 123, L 124/. The STADIF program describes the emission path for fission gases such as Kr, Xe and J by solving the stationary double phase diffusion equation. The SLIPPER program calculates the emission of the solid materials by solving the instantaneous single-phase diffusion equation. The applied diffusion parameters are taken from the HBK standard data set /L 164/ or the report / L 122/. Both programs have been verified in experiments. Also, in s subsequent evaluation by AVR, the measured cooling gas activities were compared to those computed with STATIF for the HRT Module. The computed R/B values for the HTR module are of the same order of magnitude as the measured values of AVR.

In summary, in our evaluation of the emissions rates calculated on the fuel element surfaces by the applicant we have arrived at the following results:

- The classification of the source energy levels into production-related and radiation-related particle fragmentation as well as into matrix contamination and the derivation of various emission models have been reasonably selected.
- The physical models used in the calculations represent the current state of the art and describe the essential transport procedures for fission and activation products.

- The input data used in the calculations such as, for example, the particle defect fractions, the diffusion coefficients, and other material properties are sufficiently conservative.
- In the selection of the observed nuclide vector are taken into consideration the essential, that is, the isotopes which are relevant from the radiological point of view.

Our own calculations of the activity inventory with conservative evaluations of the emitted fractions provide in some nuclides distinctly lower emission rates than those provided by the applicant; the emission rates of the other nuclides are of the same order of magnitude as those presented by the applicant. We expect, therefore that, for the operation in accordance with the specifications, the emission rates will not be higher than those given by the applicant.

Radioactive Materials Carried by the Gas in the Primary Circuit

The radioactive materials emitted from the fuel elements in the primary cooling agent, or the radioactive materials, which are formed there, are transported from the core area in the primary circuit.

In the normal operation, a quasi stationary cooling agent activity takes place, which can be calculated on the basis of a balancing, while taking under consideration all the source and loss terms.

The emission from fuel elements, which is here essentially taken into account as a source energy level, has already been discussed in the previous section. An additional source of aerosol-bonded radionuclides, which are, however, only of subordinate importance in comparison with the emissions from the fuel elements, results from the disintegration of short-lived fission noble gases carried by the primary gas.

To the loss terms belong essentially the radioactive decay, the separation in the helium purification plant, and the removal with the primary circuit leakage as well as, in the aerosol-bonded fission products, the deposit on the surface structures of the primary circuit. The main drain of the fission noble gases is formed by the helium purification plant and, in the case of iodine and the aerosol-bonded radionuclides, the metallic surface structures of the primary circuit.

To evaluate the data presented in the safety report with a view to the interpretation values for the primary gas activities, we have thoroughly checked the data presented by the applicant, and have in part carried out our own calculations.

In case of the fission noble gases, the applicant has taken into account as loss terms the radioactive disintegration and separation in the helium purification plant. The interpretation value for the purification constant amounts, according to the safety report, to 0.05 per hour. We do not see any problems from the point of view of the technical implementation of this interpretation value. The loss term via the emission with the primary gas leakage was not been taken into account, since the time constant for the leakage of 0.001 per day is negligible in comparison with the purification constant of the helium purification plant of 0.05 per hour.

The emission rates of the fission noble gas from the fuel elements provides the noble gas primary cooling agent activity as presented in Table 3.3-1.

The fission products which are emitted during the operation time of the reactor or which are produced by the disintegration of the short-lived noble gases are stored primarily on the metallic surfaces of the primary circuit, especially the steam generator, are absorbed on the reflector graphite, or are bonded to the graphite dust, and therefore accumulate in the dead rooms of the primary circuit. The metallic fission product and iodine activity contributed by the primary gas is therefore low in comparison with the surface activity.

To determine the design activity in the primary gas, the applicant has applied for all solid materials and iodine an overall accumulation rate of ten percent per circulation /U 1, U 3.3-1/. For an output of cooling agent of approximately 85 kg/s and a circulating cooling agent mass of 1,450 Kg /U 1/ is computed a plateout constant of $6.2 \times 10^{-3}\ s^{-1}$ for the evaluation case. This value can be considered to be conservative.

Under the above assumptions, the design parameter, which result for the cooling gas activities for metallic fission products, are given in Table 3.3-2 and for the iodine in Table 3.3-3.

From the noble gas followers the applicant took into account only the nuclides Rb 88, Sr 90, and Cs 138. Since the other noble gas followers are also on the level of the primary gas activity and their radioactive half-life does not differ much from the above given radio-nuclides, we consider it practical for reasons of integrity to also take into consideration the other noble gas followers. The primary gas activity of the noble gas successors is presented in Table 3.3-4. The contribution of Sr 90 from the disintegration of Kr 90 was here not explicitly presented, but is included in the value of Sr 90 in Table 3.3-2.

The C 14 emitted into the primary cooling agent is separated in the gas purifying plant and, as a result of the cyclic regeneration, it is emitted predominantly as CO_2 through the ventilation chimney into the environment. At equilibrium, the emission rate from the fuel elements is equal to the average emission rate.

For the calculation of the design activity of C 14 in the primary gas, which mainly formed by neutron recovery in the nitrogen pollutants of the fuel element graphite and arrives via corrosion in the cooling gas, were used experiments carried out by AVR in Jülich.

On the basis of the yearly emission rates at AVR, the expected emission rates in the HTR module were calculated while taking into account higher specific activity in the fuel element graphite of the HTR module and various specifications for the CO concentration in the primary helium of the AVR and of the module /U 3.3-1/. Accordingly, a yearly emission of 24 to 140 GBq per full capacity year (VLa) is expected for the HTR Model.

Because of insecurities in the individual extrapolations between AVR and module, a distinctly higher yearly emission of 520 GBq per full capacity year was taken into account for the design case. Based on these emission rates, the primary cooling agent activity C 14 is calculated to be 1.2 x 10^9 Bq.

Tritium is produced essentially during the He3 activation of the primary helium, the ternary fission, as well as during the activation of lithium in the graphite of the core components and as a result of combustion of the boron found in the absorber systems. The applicant calculates for the design case a conservative tritium production rate of 241.6 TBq per full capacity year in the warm-up phase and 95.9 TBq per full capacity year in the equilibrium core. The assumptions on which the values have been calculated are listed in the document /U 3.3-1/. To determine the primary circuit activity in the design case it was started from a conservative peak value of the emission rate of 49 Tbq per full capacity year per module, which lies with a factor of 1.6 above the equilibrium core (31 TBq per full capacity year per module). If the radioactive disintegration is ignored; the equilibrium cooling agent activity for tritium is therefore computed by 1.1×10^{11} Bq.

Accumulation of Radioactive Materials in the Primary Circuit

The source energy levels for the surface activity are the fission products emitted from the fuel elements, which accumulate by adsorption on the metallic surfaces of the primary circuit, especially the steam generator, and on the reflector graphite, as well as the dust created by abrasion and carbonization, which accumulates especially in the dead rooms of the primary circuit.

We have examined the assumptions used by the applicant as basis for determining the design parameters for the surface activities and have carried out in part our own evaluation.

In the calculations of the fission product surface activities, the applicant used as basis an overall accumulation per cycle at a level of 90 percent for the design case and 10 percent for the expected case. The relevant plateout constants amount to $6.2 \times 10^{-3}\ s^{-1}$ for the expected case and to $0.136\ s^{-1}$ for the design case /U 1/.

The accumulation rates used by the applicant are conservative since they cover the maximum and minimum value for the accumulation rates of the entire spectrum of accumulation rates observed so far in the operation of operating gas cooled reactors, for example, AVR, Dragon, Peach Bottom, Fort St. Vrain.

The summary of the data concerning the accumulation rates of fission products measured thus far in the primary circuit of gas cooled reactors is presented in one document /L 119/. Accordingly, for the AVR of experiments with Cs, Ag and J were derived accumulations of 80 to 95 percent per cycle and, based on this, a value of 30 per cent was assessed per circulation for Sr. The other reactors show a somewhat lower accumulation per cycle of Cs and Ag (50 to 80 per cent) and a clearly lower value of J (10 to 40 percent). These lower values are confirmed at AVR by new analyses of Cs (50 to 60 percent). A value of 30 percent per cycle for Sr and a value of 20 percent per cycle for iodine are viewed as conservative.

On the basis of a plateout constant of $0.136\ s^{-1}$ result the surface activities in primary circuit for design case shown in Table 3.3-5.

A surface contamination of the primary circuit by the accumulation of contaminated dust as a consequence of the abrasion of the fuel element surface was not explicitly mentioned.

Starting from the expected dust quantity in the primary circuit according to 32 VLa of a maximum of 100 Kg/ U 3.3-2/ and based on the activity concentration on the fuel element surface /U 3.3-2/ were calculated the dust-bonded surface activities listed in Table 3.3-6, while ignoring the radioactive disintegration. A comparison with the surface activities listed in Table 3.3-5 shows that the contribution of the dust-bonded activity can be ignored.

3.3.2 Radioactive Materials in the Ancillary and Secondary Systems

Helium Purification Plant

The helium purification plant has a triple strand construction with one cleaning strand being subordinated to each module unit. The third purification strand serves as a reserve in the case that one purification strand fails, or is used during the regeneration operation of a purification strand and during repair work. Each purification strand is designed in two stages and consists of

- A dust filter for the separation of metallic fission products and dust-bonded activities,

- A molecular sieve for the adsorption of tritium, chemically bonded in the form of HTO, for the adsorption of C 14 chemically bonded in the form of CO_2, as well as for the delay of fission noble gases,

- A deep temperature absorber for the adsorption of the fission noble gases as well as of the tritium and the C 14 chemically bonded in the form of CH_4.

The specific standstill of the purification plant between two regeneration cycles is 1,000 hours. Approximately 24 hours are estimated for the regeneration /U 1/. For further explanations concerning the helium purification plant see the Chapter 2.7.1.

The tritium (HTO), which accumulates during the regeneration of the molecular sieve is extracted from the regeneration gas by condensation and temporarily stored in the plant in the container for the condensate occurring until its disposal. The tritium contained in the regenerate of the deep temperature absorber in the form of CH_{3T} is transferred into HTO by catalytic combustion of methane, is then extracted by condensation and also temporarily stored in the container for the condensate the is produced during the operation.

The fission noble gas accumulated during regeneration, which arrive in the form of C 14 bonded to CO_2, as well as inseparable tritium, go into the storage container for radioactively contaminated helium and are carried away after a temporary storage unfiltered through the ventilation chimney.

The activity inventories in one strand of the helium purification plant shown in Table 3-7 have been conservatively established on the basis of the design activities in the primary cooling agent (Table 3.3-1 to 3.3-4) and the specific purification constants of 0.05 per hour, while taking into consideration the radioactive disintegration during the standstill of the purification plant between two regeneration cycles of 1,000 hours for noble gases, tritium and C 14, or during 32 TBq per full capacity year for aerosols. The retention factor in the cleaning plant has been estimated for all nuclides at 100 per cent.

The distribution of radionuclides in the individual components of the cleaning strand is presented in Table 3.3-8.

The activity inventory accumulated on the dust filters is to be considered as the maximum value according to 32 TBq per full capacity year. The highest contribution of 5.8×10^9 Bq is delivered by the fission products Sr 90, Cs 134, Cs 137, and Ag 110 m. The contributions of iodine are lower by a factor of 5 and the noble gas successor by a factor of 10.

The following delay times in the molecular sieve have been determined for the distribution of the activity inventories of the fission noble gases in the components of the molecular sieve/deep temperature absorber for Xe and Kr isotopes /U 3.3-1/:

- Kr: 0.09 h,
- Xe: 0.45 h.

In this way is calculated a total activity in the low temperature absorber which is higher by a factor of 200 in comparison with the molecular sieve. The contribution of the short-lived fission gases based on a standstill time of 1,000 hours is negligible.

When calculating the saturation activities for tritium it was assumed that the tritium present in HT form would be separated to 100 % in the molecular sieve and the tritium in CH_3T form would be separated in the low temperature absorber. According to the primary relation HT:CH_3T = 5:1 based on the AVR operational experience /U 3.3-1/, a saturation activity of 4.6 x 10^{-12} Bq in the molecular sieve and of 9.3 x 10^{11} in the low temperature absorber was calculated.

A distribution of the C 14 activity to both components of the purification plant according to the primary relation CO_2/CH_4 was not carried out; it was rather assumed that the whole 100 % amount of C 14 is retained in the molecular sieve. We do not have any objections with respect to this procedure, as this does not play a role in the emission of C 14.

The activity inventories of the short-lived noble gas followers were calculated on the basis of the saturation activities of the parent nuclides in the molecular sieve or in the deep temperature absorber from the transmission probability parent nuclide/daughter nuclide and refer to the plant standstill time of 32 full load operation years. The values given in Table 3.3-7 should be considered as maximum values.

The activity inventory in the helium purification plant was calculated on the basis of the design parameters in the primary cooling agent and thus represents the highest estimate. The distance to the expected activity inventories with the factors:

- Noble gases:	factor 3.3
- Iodine:	factor ≥ 80
- Long-lived solid substance:	factor 3.2
- H 3:	factor 1.6
- H 14:	factor 10
- Noble gas follower (without Sr 90):	factor 3.2

has been chosen sufficiently large so that no problems concerning the exceeding of the activity are expected.

According to the design, the purification strand will be regenerated every 1,000 hours. Concerning the standstill time, we do not expect any technical difficulties.

Evacuation System of the Helium Purification Plant

The evacuation system of the helium purification plant has the task of creating the necessary vacuum in the pressure relief containers during the regeneration of the purification plant after the decompression of the helium circuit needed to regenerate the molecular sieve and the low temperature absorber.

The connection of the evacuation system in the molecular sieve is carried out in the third phase of the regeneration, that is, after the expulsion of CO_2 and H_2O, and the separation of H_2O in the water cooler to remove the water and the carbon dioxide still remaining in the molecular sieve up to the minimum residues by reducing the pressure from 1 bar to 1 mbar.

The evacuation system is connected to the deep temperature absorber to remove pollutants, mainly nitrogen, xenon and krypton, from the absorber to maintain the pressure at 0.9 bar and to improve the non-absorption.

The evacuation gases are pumped into the container for radioactive contaminated helium.

The steps of the procedure for the regeneration of the components of the cleaning plant as well as the details concerning the evacuation system are described in detail in the documentation /U 2.7-1/.

The amount of noble gases produced during the regeneration, the C 14 and to a smaller extent also the tritium, are transported via the evacuation system to the storage container for the radioactively contaminated helium.

The tritium fraction entering the storage container together with the regenerate, amounts in accordance with the design to 2 % of the overall tritium inventory in the purification strand /U 3.3-1/. The value is based on AVR operational experiences and can be considered the maximum value.

The solid substances that reach the storage container can be neglected, since both the molecular sieve regenerate and the low temperature absorber regenerate are passed through filters.

The whole activity inventory needed for one regeneration cycle in the container for radioactively contaminated helium is presented in Table 3.3-9. A decay time of 12 hours

Was taken into consideration. The values are valid as maximum values are higher for the noble gases by a factor of 3.4, for the tritium by a factor 1.6, and for the C 14 by a factor 10 than the expected values.

Evacuation System for the Handling of the Fuel Elements

From the viewpoint of the system technology, the evacuation system of the operation of the fuel elements is assigned to the fuel elements loading plant and helps to evacuate the charging room for fresh fuel elements after their loading, and in this way suppresses the access of air into the primary circuit.

The evacuation air is purified by suspended particulate air filters of the system and subsequently conducted to the exhaust air chimney.

The activity derivation through the evacuation system of the fuel elements operation is not shown separately.

Based on the assumption that the amount of dust in the fuel elements loading plant expected in the documentation /U 3.3-2/ amounts to 1,000 Kg after 32 full load operation years and it if is considered that the fraction of the fresh fuel elements is approximately 7 % of the entirety of circulated partially combusted fuel elements, approximately 2 Kg of dust per year arise from the abrasive wear on the fresh fuel elements. If is assumed that the free uranium fraction in the fuel-free layer of the fuel elements ball is $\geq 6 \times 10^{-5}$, the uranium concentration amounts to approximately 9.2×10^{-6} g uranium per 1 gram of dust. On the basis of a retention factor of 10^{-4}, which can be normally performed with suspended particulate air filters, the yearly discharge of the dust-bonded activity through the exhaust chimney is less than 10^3 Bq per module. This contribution can be neglected in comparison with the other removals.

Evacuation System of the Primary Circuit

The task of the system is to evacuate the whole primary circuit and the accessory systems filled with helium, after loading the core with fuel elements and before filling it with helium, to bring to the specific concentrations the gaseous pollutants, especially nitrogen.

During the recurring tests and maintenance and repair work, the space between the core container and the reactor pressure vessel above the core container support is evacuated before opening and after closing after the pressure relief of the primary circuit. The evacuation system is equipped with the system's own filters so that the carrying in of dust-bonded activities is minimized.

The helium to be conducted away before opening is conveyed via the pressure relief system of the helium accessory circuits and of the fuel elements handling, whereby an emission of the radioactive materials is prevented.

After closure, the air is removed from the primary circuit via the evacuation system. The evacuation air is conducted to the chimney through the exhaust air-filtering device connected during maintenance and repair work.

The applicant is not expecting the release of a substantial amount of activity.

> We have examined the statements of the applicant concerning the expected activity removal and we can confirm them.

As a result of the measures for reducing the activity release during maintenance and repair work on contaminated components of the primary circuit stated in the documentation of the applicant /U 3.2-1/ which are

- The removal of the greatest part of the cooling agent activity borne by the primary gas by pressure relief,
- The interruption of the connection of the part of the reactor pressure vessel not flown through to the primary circuit by closing the stop devices in the pressure differential valve,
- The evidence of the sufficient tightness of the inner primary circuit closure by means of a tightness test according to the pressure increase method,
- The evacuation of the reactor pressure vessel not flown through via the evacuation system,
- Work on the contaminated systems or their dismantling according to the plastic bag method or under tents,
- The filtering of the evacuation stream via dust filters,
- The filtering of the exhaust air via the exhaust air filtering plant.

as well as taking into consideration that only stagnant gas is present in the respective section of the primary circuit and therefore the surface contamination is low, the evacuation system of the primary circuit can be neglected as a separate load path. The low emissions are sufficiently covered by the other load paths.

Pressure Release System of the Helium Ancillary Circuits and the Fuel Element Handling

All systems from which helium has to be removed because of operational requirements are attached to the pressure release system. The most important systems are:

- The helium purification plant,
- The evacuation system for the primary circuit and the helium purification plant,
- The fuel elements handling arrangements,
- The sample taking system (gas analysis system),
- The water removal system for the helium accessory systems.

The system is designed so that it can absorb all decompression gases simultaneously. Depending upon their chemical or radioactive pollution, the decompression gases are transported back either

- Into the storage container for radioactively contaminated helium, or
- To a location in front of the helium purification plant.

The limit values for the activity concentrations based on which it is decided as to the permanence of the decompression gases, have not been set yet. This issue needs detailed analysis within the course of the design development. However, in general we do not see any problems concerning the limitation of the activity removal through the exhaust air chimney according to the design parameters since there is always the possibility of a previous purification of the decompression gases.

The highest contribution of the activity release in the storage container supplies the pressure relief of the helium purifying plant before its regeneration. In this way, on the basis of the design activities in the primary circuit, approximately 10^{11} Bq of nobles gases, 1.2×10^{10} Bq of tritium, and 1.3×10^{8} Bq of C 14 enter the storage container. In the noble gases, the contribution of the short-lived radionuclides is dominant. Aerosols and iodine can by neglected.

Possible aerosol inputs via the evacuation systems for the primary circuit and the fuel elements handling can also be neglected because the evacuation air is filtered.

If it is assumed that the entire activity inventory of the storage container of the pressure relief is also conducted after the regeneration of the helium purification plant to the storage container for radioactive polluted helium, then this activity contribution is contained in the activity inventory shown in Table 3.3-9.

Nuclear and Safeguarded Intermediate Cooling System

The following cooling system components are connected to the intermediate cooling system:

- The primary circuit blower,
- The cooler and blower of the helium purification plant,
- The fuel elements loading and unloading apparatus,
- The wastewater system,
- The surface cooler,
- The support of the pressure vessel unit,
- The nozzles on the reactor pressure vessel floor.

Deionised water is used as the cooling agent in both systems. To prevent corrosion, the oxygen quantity is limited.

The activity in the systems arises from the activation of the structural materials and the subsequent corrosion, from the activation of water and its contaminants, as well as from leakage in the primary cooling agent-conducting system.

The intermediate cooling systems are operated as closed cooling systems. A removal of sludge from the circuits is not planned. As a result, activity releases from the systems occur only when there is a leakage and the containers are drained. The expected maximum activity inventory in the systems amounts to 3 x 10^8 Bq /U1/. In principle, there exists the possibility of checking the activity in the circuits by taking samples.

The activity in the cycles of the intermediate cooling systems is determined by the activation of the structural materials in the area of the surface coolers and the subsequent corrosion as well as by the activation of the cooling agent and its contamination.

Using the experiences with reactors already in operation, the corrosion in the circuits and the generation of active products can be influenced to a certain extent by selecting appropriate materials and using the appropriate conditioning.

The expected activity concentrations in the area of the surface cooler resulting from the activation was assessed for 32 full load operation years of the reactor /U 3.3-5/. In principle, the activation products Cr 51, Mn 54, Fe 59, Co 60 und Ta 182 are concerned.

To minimize the corrosion, the oxygen in the cooling agent is limited so that the corrosion rate of 10^{-3} mm/a used as a basis in the documentation of the applicant /U 1/ can be considered a technically feasible value.

Applying the activity concentrations in the surface cooler stated in /U 3.3-5/ and the above-mentioned corrosion rate, the activity inventory shown in Table 3.3-10 lies in the intermediate cooling circuits at the end of the reactor operation time of 32 full load years. With the values calculated using conservative assumptions, Co 60 with 3 x 10^{-8} Bq delivers the essential activity contribution. The contribution of the Ta 182 is lower approximately by a factor of 10 and the contribution of the other nuclides is lower by at least a factor of 100.

The additional activity inputs into the safeguarded intermediate cooling system resulting from primary cooling agent leaks do not arise, as the system is not directly connected to the primary circuit. The activity of the individual strands is monitored by regular sample taking.

In the nuclear intermediate cooling system exists the possibility of activity penetration from the primary circuit resulting from leakage. How high this contribution will be in particular depends on the tightness of the systems and cannot be estimated in advance. However, by regular sample taking, leakages can be discovered early enough in order to implement the necessary measures and reduce the activity penetrations to a minimum.

Because both systems are operated as closed circuits, activity releases from both systems are possible only when leakage occurs or when the containers are drained. The radioactive materials released when there is leakage or container drainages are sufficiently covered by the design parameter for the removal with wastewater.

Nuclear and Safeguarded Ancillary Cooling Water System

The nuclear and safeguarded intermediate cooling systems transfer the absorbed warmth in the intermediate coolers to the respective accessory cooling water systems.

For the re-cooling of the nuclear ancillary cooling water flow is available a hybrid cooling tower.

Several wet cooling cells provide the re-cooling of the safeguarded ancillary cooling water flow.

The ancillary cool water systems are normally activity-free. However, as a result of the structuring of the pressure, leakage and subsequent activity penetrations into the ancillary cooling water systems resulting from un-tightness in the intermediate coolers may occur.

To check the tightness of the intermediate coolers, both circuits are continuously monitored with a gamma-sensitive detector as well as by regular sample taking – wherein the safeguarded ancillary cooling water system is monitored from the collecting line and in the nuclear ancillary cooling water system in the supply line to the hybrid cooling tower - /U 1/.

The measures are in our opinion sufficient to recognize the possible activity penetrations resulting from leakage early enough to implement the necessary rectifying measures.

Water Vapor Circuit

As a consequence of the pressure difference in the direction of the primary circuit, activity penetrations possible at the steam generator are not possible either during the full-scale operation or during the starting and shut down phases. Therefore activity development in the water steam circuit occurs only via the creation of activation products in the area of the steam generator and as a consequence of tritium permeation from the primary circuit.

A continuous purification of the water steam circuit is not planned, among other reasons, because of the expected low activity concentrations. The operation of the condensate purifying plant during full-scale operation is planned only when a condenser leakage occurs, which is to be expected only seldom because of the material selected for the condenser.

The tritium permeation through both steam generators is of 8.4×10^7 Bq/h according to the safety report. In the closed operational mode, and with an assumed cooling agent loss rate on the secondary side of 1.25 Mg/h, the resulting tritium saturation activity concentration in the water steam circuit is of 6.3×10^7 Bq/Mg.

The design parameter of activity concentration in the fresh steam for the activation products arising from the activation of the trace pollution of the feed water amounts to 0.4 Bq/Mg.

The tritium permeation through the steam generator surface depends on the pressure gradients, the temperature conditions, as well as on the form in which the tritium is present. The oxide cover layers on the steam generator surfaces have a permeation-inhibiting effect on the vapor generator surfaces, which causes a reduction of the activity penetration. In

Shutdown operations, the permeation-inhibiting effect of the oxide layer is weakened by the formation of cracks. This leads, until the layer recovers, to an activity concentration higher by a factor of 2 or 3 during the starting mode /L 120/.

While tritium in the HT form permeates directly, HTO and CH_3T can permeate into the water steam circuit only via oxidation and carbonizing processes. Accordingly, the penetration rate of the two latter is substantially lower.

For the stationary operating mode is set a permeation rate of 4.2×10^7 Bq/h per module based on the design parameter for tritium. Therefore, the fact that in the primary circuit one part of the tritium is present in the form of $CH_3T$was neglected because of the conservative approach. In the secondary circuit, an even distribution of the tritium over the water and steam phase is carried out so that the balance activity in the secondary circuit during the closed operational mode is determined by leakage losses suffered on the secondary side.

A value of 1.25 Mg/h, derived from the operational dates of the THTR /U 3.3-3/ was estimated for these losses. Basically we do not see any difficulties in complying with the leakage rates, since the operation of the latest light water reactors has shown that this value is technically feasible.

On the basis of the above-mentioned tritium permeation and the leakage rate, the tritium activity concentration in the secondary circuit amounts to 6.3×10^7 Bq/Mg in the closed operational mode. In this value, the short-term higher tritium penetration rates during the starting operations are also covered.

The design parameter for the activation products arising from the activation of the trace contaminants in the feed water was determined using essentially the values measured in the AVR in the fresh steam while taking into consideration the different thermal neutron flows at the entrance of the steam generators /3.3-4/. The concentrations measured in the AVR are, also in the stationary operational mode, subject to certain fluctuations which, depending on the selected proof method, can amount to up to a factor of 100. The design parameters for the concentration in the HTR module shown in Table 3.3-11 were calculated with the exception of Na 24 from the maximum values measured using the long-term filtration in the years 1974 – 1978 in the AVR. For Na 24, the mean value from measurements before 1974 was used.

The design parameter for the activity concentration of the activation products in the fresh steam amounting to 0.4 Bq/Mg was determined according to a conservative approach and is with the exception of Na 24 clearly over the mean values. The conservative character of the design parameter is confirmed by the measurement results in THTR /U 3.3-4/.

Because of the low activities in the fresh steam, an activity check by taking samples is carried out. With regard to the fact that a part of the fresh steam is transported directly away as process steam, the method of activity monitoring – a continuous monitoring or a taking of samples – is to be discussed in detail during the construction planning.

Process Steam System

In principle, it is possible to operate the secondary circuit in the open mode and to divert one part of the fresh steam as process steam. Accordingly, the activity concentrations in the fresh steam are lower.

The applicants has the intention of applying the following values within the framework of a later approval procedure for the tritium activity concentrations in the process steam:

- Stationary operation: 5.0×10^6 Bq/Mg
- Start-up operation: 7.4×10^7 Bq/Mg in max. 20 days per year

A monitoring should be carried out by taking samples /U 1/.

We do not see any problems concerning the compliance with these values. The difference with the design parameters for the tritium concentrations in fresh steam during the open operation mode is with a factor of 10 for the stationary operation and a factor of 2 for the short-term start-up operation large enough.

Both values meet the demands of § 4, Section 2 of the Radiation Protection Regulation /L 2/ allowing in combination with Supplement III, Section 2, an approval-free operation with the radioactive materials, should the specific activity be lower than 74 Bq/g.

By utilizing the process steam outside of the power plant and thus beyond the controlled area, unlimited further utilization of the process steam condensate must be guaranteed. Should no yearly limits for the drawing off of tritium through the process steam path during the later approval procedure be set, § 4, Section 2 of the Radiation Protection Regulation must be complied with, allowing the transportation of the process steam condensate into the wastewater channel or into the surface waters only if the activity in the wastewater does not exceed a 1.25 multiple of the value stated in Supplement IV, Table IV 1, Column 6.

If it cannot be generally excluded that the condensate will be used as feed water, the maximum yearly supply of tritium set by § 46 may be slightly exceeded – based on the intended application values, assuming a yearly feed water consumption of 880 l /L 13/. For this reason, this aspect has to be set before the approved values are finally examined.

During the course of the design development, it must be additionally examined, whether additional requirements under § 4 Section 4, Point 2 b-d, of the Radiation Protection Regulation are to be applied during the utilization of the process steam.

The evidence that the intended application values are complied with has to be produced by taking samples. Whether this way of checking is sufficient or whether continuous checking is necessary depends on the utilization of the process steam and must be decided during the course of the later approval procedure.

3.3.3 Radioactive Materials in the Ambient Air of the Reactor Cell

The probable sources of radioactive materials in the reactor cell air are in principle the following paths:

- The activation of the radioactive materials in the ambient air of the reactor pressure vessel as source of Ar 41,
- The activation of the metallic surfaces on the outer side of the reactor pressure vessel and of the surface cooler as a source of aerosols,
- The primary cooling agent leakage as a source of aerosols and noble gases.

The reactor cell is separated from the neighboring rooms. The design parameter for the leakage rate from the neighboring cells into the reactor cell amounts according to the safety report to 2, 000 m^3/d. To preserve an underpressure of 1 to 1.5 mbar, approximately 2, 000 m^3 per day (free air volume of the cell) are drawn off and removed through the exhaust air chimney.

Activation of the Ambient Air

The components existing in the air – nitrogen, carbon, oxygen, and argon – are activated by the neutron flow prevailing in the cell.

The most important reactions are:

- Ar	40	(n, γ)	Ar	41
- O	16	(n, p)	N	16
- O	18	(n, γ)	O	19
- N	14	(n, p)	C	14

Argon 41 makes the highest contribution to the air activity. Based on the two-dimensional neutron flow fields, the applicant calculated the macroscopic reaction times at the edges of the air gap between the reactor pressure vessel and the cell wall according to the mesh points method and a production rate of 1.1×10^6 Bq/s was calculated by integration of the macroscopic reaction rates over the volume of the cell area concerned /U 3.3-1/. The basis for the design case is higher by the factor 2 than the production rate. With regard to the air renewal rate of $1,16 \times 10^{-5}\ s^{-1}$, a balance activity of 1.9×10^{10} Bq in the reactor cell is the resulting value. Our own estimates confirm the conservative character of these values.

The applicant did not consider other activation products. Under the aspect of direct radiation, we have additionally evaluated the N 16 created via a (n, p) process from O 16. However, because this process is in progress with neutron energies of more than 9 MeV, the generation rate of N 16 and consequently the activity can be neglected.

Aerosol Concentration in the Ambient Air

The following are the most important sources for aerosol contamination in the ambient air:

- The creation of activation products on the metallic surfaces of the reactor pressure vessel and the surface cooler by the neutron flow coming out of the reactor pressure vessel and the abrasive wear of the surfaces. In principle, the most important activation products are Cr 51, Mn 54, Mn 56, Fe 59, Co 58, Co 60, and Ta 182.

- The aerosol release because of helium leakages. The aerosol spectrum released corresponds to the spectrum in the primary circuit.

The applicant considered only the aerosol activity produced by the activation products.

We have estimated the contribution of the aerosol concentration in the primary cell by helium leakages according to a conservative approach. It was assumed that the specific helium leakage of 1 %o per day was released only in the primary cell. If we consider only the long-lived aerosols with $T_{1/2} = 8$ days, the result is a balanced concentration of 0.45 Bq/m^3 with the nuclide-specific rates shown in Table 3.3-12 in the cell. The contribution can be neglected in comparison with the activation products.

The applicant assessed the possible aerosol contributions by the activation products on the basis of two possible models /U 3.3-1/. The basis for the activation concentrations of the activation products in the reactor pressure vessel and in the surface cooler are the values for a 32 year reactor full-load operation shown in the documentation /U 3.3-5/. While considering the activation products Co 60, Fe 59, and Ta 182, the aerosol-bound activity concentration in the cell is between ≅ 2 Bq/m^3 and 70 Bq/m^3 or, considering all the activation products with $T_{1/2} = 8$ days, it is between ≅ 2,8 Bq/m^3 and 90 Bq/m^3. With regard to the conservative assumptions for estimating the possible activity concentration in the ambient air and based on the assumption that if required, the exhaust air can also be drawn off after filtration, we do not see any problems in meeting the design parameter. The nuclide-specific composition of the aerosol-bonded dust activity with respect to all activation products with $T_{1/2} = 8$ days is shown in Table 3.3-13. For the radiological calculation, the applicant selected Co 60 as the representative nuclide for the activation products. We agree with this procedure, since by the selection of Co 60, the other radionuclides are also covered.

Fission Noble Gas Concentration in the Ambient Air

The fission noble gas concentration in the ambient air of the primary cell is not explicitly stated in the safety report. Based on the assumption that the entire primary cooling agent leakage occurs in the cell area, the resulting balance concentration of 1.5×10^5 Bq/m^3 arises in accordance with the nuclide-specific survey in Table 3.13-14. The contributions of the noble gas followers, especially of Sr 90 caused by the decay of Kr 90, can be neglected. The values indicated here should be considered the highest maximum, since in reality the greatest fraction of the helium leakage can be expected outside the reactor cell.

Table 3.3-1: Stationary noble gas cooling agent activity in the HTR module

Radionuclide	Primary Cooling Agent Activity in Bq
Kr 83 m	$1.9\ 10^{10}$
Kr 85 m	$5.8\ 10^{10}$
Kr 85	$3.2\ 10^{8}$
Kr 87	$7.2\ 10^{10}$
Kr 88	$1.4\ 10^{11}$
Kr 89	$2.8\ 10^{10}$
Kr 90	$1.2\ 10^{10}$
Xe 131 m	$1.2\ 10^{9}$
Xe 133 m	$1.1\ 10^{10}$
Xe 133	$2.3\ 10^{11}$
Xe 135 m	$1.7\ 10^{10}$
Xe 135	$1.4\ 10^{11}$
Xe 137	$4.8\ 10^{10}$
Xe 138	$9.1\ 10^{10}$
Xe 139	$1.5\ 10^{10}$
Total noble gases	$8,8\ 10^{11}$

Table 3.3-2: Stationary solid cooling agent activity in the HTR module

Radionuclide	Primary Cooling Agent Activity in Bq	
Cs 134	2.6	10^5
Cs 137	5.4	10^5
Ag 110 m	1.9	10^4
Sr 90	2.1	10^3
Total solids	8.2	10^5

Table 3.3-3: Stationary iodine cooling agent activity in the HTR module

Radionuclide	Primary Cooling Agent Activity in Bq	
J 131	3.4	10^7
J 132	4.5	10^8
J 133	2.2	10^8
J 134	1.1	10^9
J 135	3.9	10^8
Total iodine	2.2	10^9

Table 3.3-4: Stationary noble gas follower cooling agent activity in HTR module

Radionuclide	Primary Cooling Agent Activity in Bq	
Rb 88	1.3	10^{10}1
Rb 89	3.0	10^{9}
Rb 90	5.0	10^{9}
Sr 89	7.6	10^{4}
Cs 138	5.0	10^{9}
Cs 139	2.5	10^{9}
Ba 139	5.4	10^{7}
Total n. gas follower	2.8	10^{10}

Table 3.3-5: Sedimentary activity on the primary circuit surfaces according to 32 FY (VLa – full load operation years)

Radionuclide	Surface Activity in Bq
I 131	2.1 10^{11}
I 132	3.3 10^{10}
I 133	1.5 10^{11}
I 134	3.2 10^{10}
I 135	8.5 10^{10}
Total iodine	5.1 10^{11}
Rb 88	1.2 10^{11}1
Rb 89	2.5 10^{10}
Rb 90	7.0 10^{9}
Sr 89	2.9 10^{9}
Cs 138	8.5 10^{10}
Cs 139	1.2 10^{10}
Ba 139	2.4 10^{9}
Total noble.gas follower	2.5 10^{11}
Sr 90	9.3 10^{9}
Cs 134	1.5 10^{11}
Cs 137	2.4 10^{12}
Ba 110	3.7 10^{9}
Total long-lived solids	2.6 10^{12}

Table 3.3-6: Dust-bound surface activity on the primary circuit surfaces acc. to FY (VLa – full load operation years)

Radionuclide	Dust-bonded Surface Activity in Bq
I 131	8.9 10^8
Cs 137	7.0 10^7
Sr 90	5.6 10^7
Ag 110 m	1.3 10^7
Total	1.0 10^9

Table 3.3-7: Activity inventory in a helium-cleaning strand at the end of a lifetime of 1,000 h or 32 FY (La – full load operation years)

Radionuclide	Total Activity in Bq in a Purification Strand	
Kr 83 m	2.5	10^{9}
Kr 85 m	1.9	10^{10}
Kr 85	1.6	10^{10}
Kr 87	6.6	10^{9}
Kr 88	2.7	10^{10}
Kr 89	1.1	10^{8}
Kr 90	7.6	10^{6}
Xe 131 m	2.2	10^{10}
Xe 133 m	4.4	10^{10}
Xe 133	2.1	10^{12}
Xe 135 m	3.2	10^{8}
Xe 135	9.5	10^{10}
Xe 137	2.2	10^{8}
Xe 138	1.5	10^{9}
Xe 139	1.2	10^{7}
Total noble gases	2.3	10^{12}
J 131	4.7	10^{8}
J 132	7.7	10^{7}
J 133	3.3	10^{8}
J 134	7. 1	10^{7}
J 135	1.8	10^{8}
Total iodine	1.1	10^{9}

Continuation Table 3.3-7

Radionuclide	Total Activity in Bq in a Purification Strand	
Rb 88	2.7	10^{10}1
Rb 89	1.1	10^{8}
Rb 90	7.6	10^{6}
Sr 89	1.1	10^{8}
Cs 138	1.5	10^{9}
Cs 139	1.2	10^{9}
Ba 139	1.2	10^{9}
Total noble gas follower	3.1	10^{10}
Cs 134	3.3	10^{8}
Cs 137	5.4	10^{9}
Ag 110 m	8.2	10^{6}
Sr 90	2.1	10^{7}
Total solids	5.8	10^{9}
H 3	5.6	10^{12}
C 14	5.9	10^{10}

<u>Table 3.3-8:</u> Distribution/spread of activity inventory in helium cleaning strand at the end of a lifetime of 1,000 h or 32 FY (VLa – full load operation years)

Nuclide	Dust Filter	Molecular sieve	Deep Temp. Adsorber.
Kr 83 m		$8.3\ 10^{7}$	$2.4\ 10^{9}$
Kr 85 m		$2.6\ 10^{8}$	$1.8\ 10^{10}$
Kr 85		$1.4\ 10^{6}$	$1.6\ 10^{10}$
Kr 87		$3.2\ 10^{8}$	$6.3\ 10^{9}$
Kr 88		$6.0\ 10^{8}$	$2.7\ 10^{10}$
Kr 89		$7.4\ 10^{7}$	$3.3\ 10^{7}$
Kr 90		$7.6\ 10^{6}$	$7.3\ 10^{3}$
Xe 131 m		$2.6\ 10^{7}$	$2.2\ 10^{10}$
Xe 133 m		$2.6\ 10^{8}$	$4.3\ 10^{10}$
Xe 133		$5.2\ 10^{9}$	$2.1\ 10^{12}$
Xe 135 m		$2.3\ 10^{8}$	$9.4\ 10^{7}$
Xe 135		$3.2\ 10^{9}$	$9.1\ 10^{10}$
Xe 137		$2.2\ 10^{8}$	$1.7\ 10^{6}$
Xe 138		$1.1\ 10^{9}$	$4.1\ 10^{8}$
Xe 139		$1.2\ 10^{7}$	-
Total noble gas.		$1.2\ 10^{10}$	$2.3\ 10^{12}$
J 131	$4.7\ 10^{8}$		
J 132	$7.7\ 10^{7}$		
J 133	$3.3\ 10^{8}$		
J 134	$7.1\ 10^{7}$		
J 135	$1.8\ 10^{8}$		
Total iodine	$1.1\ 10^{9}$		

Continuation Table 3.3-8

Nuclide	Dust Filter	Molecular. Sieve	Deep Temp. Adsorber
Rb 88	$2.8\ 10^{8}$	$6.0\ 10^{8}$	$2.7\ 10^{10}$
Rb 89	$5.5\ 10^{7}$	$7.4\ 10^{7}$	$3.3\ 10^{7}$
Rb 90	$1.6\ 10^{7}$	$7.6\ 10^{6}$	$7.3\ 10^{3}$
Sr 89	$6.6\ 10^{6}$	$7.4\ 10^{7}$	$3.3\ 10^{7}$
Cs 138	$1.9\ 10^{8}$	$1.1\ 10^{9}$	$4.1\ 10^{8}$
Cs 139	$2.8\ 10^{7}$	$1.2\ 10^{7}$	-
Ba 139	$5.4\ 10^{6}$	$1.2\ 10^{7}$	-
Total n. g. f.	$5.8\ 10^{8}$	$1.9\ 10^{9}$	$2.8\ 10^{10}$
Cs 134	$3.3\ 10^{8}$		
Cs 137	$5.4\ 10^{9}$		
Ag 110 m	$8.2\ 10^{6}$		
Sr 90	$2.1\ 10^{7}$		
Total solids	$5.8\ 10^{9}$		
H 3		$4.6\ 10^{12}$	$9.3\ 10^{11}$
C 14		$5.9\ 10^{10}$	

Table 3.3-9: Activity occurrence in a storage container for radioactive contaminated helium after a regeneration cycle considering a decay time of 12 hours

Radionuclide		Total Activity in Bq in Storage Container
Kr	83 m	2.7 10^{12}
Kr	85 m	2.9 10^{9}
Kr	85	1.6 10^{10}
Kr	87	9.5 10^{6}
Kr	88	1.4 10^{9}
Kr	89	-
Kr	90	-
Xe	131m	2.2 10^{10}
Xe	133 m	3.7 10^{10}
Xe	133	2.0 10^{12}
Xe	135 m	-
Xe	135	3.8 10^{10}
Xe	137	-
Xe	138	-
Xe	139	-
Total solids		2.1 10^{12}
H 3		1.1 10^{11}
C 14		5.9 10^{10}

Table 3.3-10: Activity inventory in the nuclear and safeguarded intermediate cooling systems as a consequence of corrosion abrasion in the surface cooler after 32 full load operation years

Nuclide	Activity Inventory in Intermediate Cooling Systems in Bq
Cr 51	7.0 10^5
Mn 54	3.2 10^6
Co 60	3.0 10^8
Fe 59	3.0 10^6
Ta 182	2.0 10^7
Total	3.3 10^8

Table 3.3-11: Design parameters for the activity concentration in fresh steam during closed operational mode

Nuclides	Activity Concentration in Fresh Steam in Bq/Mg
H 3	6.7 10^7
Na 24	0.3
Mn 56	0. 05
Cr51	0. 006
Mn 54	0. 002
Fe 59	0. 04
Co 60	0. 005
Ag 110 m	≅ 0. 001
Sb 124	≅ 0. 001
Total activation products	0.4

Table 3.3-12: Aerosol-bonded activity concentration in the primary cell resulting from a He leakage

Nuclide	Activity Concentration in the Primary Cell Resulting from He Leakage in Bq/m^3
Cs 134	0.13
Cs 137	0.27
Ag 110 m	$9.5 \ 10^{-3}$
Sr 90	$1.1 \ 10^{-3}$
Total	0.41
J 131	17.0
J 132	27.5
J 133	60.0
J 134	27.0
J 135	55.0
Total iodine	186.5

Table 3.3-13: Design parameters for the activity concentrations of activation products and Ar 41 in the primary cell

Nuclide	Activity Concentration in the Primary Cell in Bq/m^3
Cr 51	1.8
Mn 54	0.7
Fe 59	4.5
Co 58	0.6
Co 60	11.3
Ta 182	11.3
Total activation products	30.2
Ar 41	$9.5\ 10^6$

Table 3.3-14: Maximum noble gas activity concentration in the primary cell resulting from He leakages

Radio-nuclide	Activity Concentration of Noble Gases in Bq/m^3	
Kr 83 m	9.5	10^2
Kr 85 m	6.0	10^3
Kr 85	1.6	10^2
Kr 87	2.6	10^3
Kr 88	1.0	10^4
Kr 89	4.5	10^1
Kr 90	3.3	10^0
Xe 131 m	5.5	10^2
Xe 133 m	4.2	10^3
Xe 133	1.0	10^5
Xe 135 m	1.3	10^2
Xe 135	2.5	10^4
Xe 137	9.0	10^1
Xe 138	6.5	10^2
Xe 139	5.0	10^0
Total noble gases	1.5	10^5

3.4 Emission of Radioactive Materials and Radiation Exposure in the Environment

3.4.1 Emission of Radioactive Materials

3.4.1.1 Emission with Wastewater

The wastewater generated in the power plant is separated according to their chemical quality and specific activity and are collected in wastewater tanks and transported after their treatment to a transfer container. Before the discharge into the receiving water, a crucial measurement according the KTA regulation 1504 /L 61/ is carried out. If the activity concentration higher than $1{,}85 \times 10^7$ Bq/m^3, the wastewater is conducted to a repeat treatment and otherwise it is discharged into the receiving water.

The different wastewaters are divided into two groups. According to the safety report, the Wastewaters Group I, the active wastewaters, contains the following:

- Sludge and leakage waters,
- Wastewaters from hot laboratories,
- Wastewaters from the decontamination room,
- Decantation water from the concentrate container of the wastewater system,
- Wastewaters from the evaporator of the wastewater system,
- Wastewaters from the concentrate treatment.

For the waters of this group, an evaporation treatment is planned /U 3.3-1/. A decontamination factor of 10^3 to 10^5 should be achieved /U 3.3-8/. A pre-purification via a centrifuge is possible.

The wastewaters of Group II

- Wastewaters from the laundry,
- Wastewaters from the shower and washrooms

are purified from mostly non-dissolved materials or, if applicable, materials with low activity.

According to the safety report, a yearly occurrence of 5 000 m^3 of wastewaters with an activity load of 3,7 x 10^6 Bq/m^3 before purification and of less than 3,7 x 10^5 Bq/m^3 after purification is calculated. The resulting yearly calculation of the discharge into the receiving water is 1,85 x 10^9 Bq/m^3.

The nuclide vector for the calculation of the radiation exposure established by us is shown in the Table 3.4-1. For the calculation of the tritium discharge caused by the secondary circuit leakage, a water leakage of 1 Mg/h has been assumed. With the design parameter for the tritium concentration in the secondary circuit of 6,3 x 10^7 Bq/m^3, the tritium discharge into the receiving water is of 5,9 x 10^{11} Bq/m^3 per year.

The estimate carried out by the applicant about the wastewater occurrence and its nuclide-specific composition can be considered only an approximation, since the operational experience from other reactors could be used only to a limited extent because of the differences existing between the plants.

As with the AVR, the wastewater is conducted to the collecting and treatment/purification plant of the KFA Jülich, and it is checked only for integral activity concentrations and tritium, no nuclide-specific evaluations are available that could be applied to the module plant. Therefore, the nuclide composition in raw water after ten years full load operation has been examined /U 3.3.-1/ at AVR on the basis of the specific dust activity. It is our opinion that the our assumptions used as a basis for finding the nuclide vector are sufficiently conservative. The alpha emitters are not explicitly considered in the nuclide vector; but they could not yet be detected in the THTR wastewaters. The conservative estimates about the possible activity contribution of the alpha emitters on the basis of the relation between the beta/alpha activity in the graphite dust of the AVR of more than 1×10^4 result in a maximum yearly alpha emitter discharge of 1×10^6 Bq. This contribution is considered when calculating the radiation exposure.

When examining the activity inventories in the components of the wastewater treatment plant stated in the documentation /U 3.3-1/, we found that a lower percentage was assumed for iodine than stated in the raw water vector. We have therefore made our own calculations for iodine. The activity inventories for iodine stated in the Tables 3.4-2 and 3.4-3 are based on an iodine percentage of 2.6 % corresponding to the percentage in the raw water. The nuclide vector for the pure water would be changed by these figures only slightly and was therefore applied without change.

3.4.1.2 Emission with Exhaust Air

3.4.1.2.1 Emission Through the Exhaust Air Chimney

The radioactive leakage through the exhaust air chimney are composed of the following contributions:

- The noble gas, iodine, C 14, H 3, and aerosol emissions caused by leakage from the primary circuit and the connected systems conducting primary cooling agents. The calculation of the yearly discharge is based on the primary cooling agent leakage of 1 %o per day and the module as well as an average air renewal rate of 1 h^{-1}.

- The noble gas, iodine, C 14, and H 3 emissions from the storage containers for radioactively contaminated helium. According to the design, 15 regenerations per year were assumed for both modules.

- The emission of the activation products with the ambient air of the primary cells. The air renewal rate amounts to $\alpha = 1\ d^{-1}$ /3-3-1/.

For all three of the emission paths was assumed a continuous and unfiltered exhaust through the exhaust chimney.

Discharge Caused by Primary Cooling Agent Leakage

When evaluating the radioactive discharges caused by primary cooling agent leaks, we assume that the leakage rate of 1 %o per day and module specified by the applicant and based on the experience obtained during the start-up of the THTR can be maintained. From comparisons with other gas cooled reactors in operation, of which all have higher leakage rates (Peach Bottom: 1 %, AVR and Dragon reactor: 0,2 %) /L 121/, high requirements concerning tightness of the components and systems result, which can, however, be technically feasible. In the design development of the components and systems this aspect will deserve special attention.

Additionally, via special reserves in the design of other components it can also be safeguarded that even if an unexpectedly high leakage occurs, the discharged radioactive materials still remain under the design parameters or value of the application.

For the calculation of the emission rates, the applicant assumes that the leakages occur in the reactor body and that the radioactive materials released in this way will be discharged according to the air renewal rate of 1 h^{-1} usual in such buildings. Possible leaks into the primary cells are not considered. With regard to the emissions, this procedure is conservative.

The yearly discharges through the exhaust chimney caused by the primary cooling emissions are for the 2nd module stated in the Table 3.4-4. The chemical form of the radioiodine is assumed to be as elementary at 100 %.

Discharge from the Storage Containers for the Radioactively Contaminated Helium

According to the design for the purification strand lifetime of 1000 hours, a total of 15 regenerations per year are required for the 2-module plant. Based on the activity inventory in the storage container occurring per regeneration indicated in the Table 3.3-9, the amount of radionuclides indicated in the Table 3.4-5 is discharged through the exhaust air chimney. The radioactive decay during the storage and the discharge procedure has been neglected applying a conservative approach.

For the calculation of the radiation exposure, the applicant assumes a continuous emission /U 3.3-6/.

In the regulation concerning § 45 of the Radiation Protection Regulation /L 13/, the following prerequisites concerning the applicability of the long-term expansion factor with uneven emissions are listed:

- The activity amount emitted within a day, that is, per 24 hours should not exceed 1/100 of the tolerated yearly emission.

- These emissions do not occur systematically at the same time of the day but are approximately evenly distributed over all day hours.

- In any time period within a half-year, the highest rate that must not be exceeded is half of the admissible yearly emission.

Based on these prerequisites, a minimum emission time of the activity inventory of a storage container of seven days was calculated in the documentation /U 3.-3-6/. When designing the system control, attention must be paid to the requirement according to which at least the emission time of seven day is to be kept and a continuous activity discharge is to be carried out within this period.

Additionally, attention has to be paid to the requirement that the regeneration of the both purification strands should not be carried out immediately one after another but with an interval of 500 hours to achieve the highest possible distribution of the activity discharges over the whole year.

Before every discharge of the container, a proof via the respective measurements/taking samples /U 3.3-6/ should be produced, safeguarding the compliance with the requirements concerning the maximum admissible per day emissions for all nuclides or nuclide groups respecting a discharge period time of at least seven days. The activity measuring points in the exhaust air chimney carry out additional checks of the total discharge.

Activity Discharge with the Ambient Air in the Primary Cell

The activity discharges through the exhaust air chimney for the 2 module-plant are indicated in the Table 3.4-6 and are based on the activity inventory in the primary cell and an air renewal rate of $\alpha = 1\ d^{-1}$ as indicated in Table 3.3-13.

3.4.1.2.2 Emissions from the Machine Housing

Through the roof ventilator of the machine housing are discharged radioactive materials because of leaks in from the water steam circuit. When calculating the yearly discharges through the roof of the machine housing, the applicant assumes a loss rate of 1,25 Mg/h on the secondary end and a loss rate of water to steam of 4:1. The assumptions are based on the operational results of the THTR /U 3.3-3/.

The yearly discharge of radioactive materials through the roof of the machine housing calculated on the basis of a steam leakage of 0,25 Mg/h are shown in Table 3.4-7.

We do not see any problems concerning the maintenance of the total leakage since the operational experience with the light water reactors shows that these values are technically feasible. However, the assessment of the operational data of the THTR /U 3.3-7/ concerning steam leaks without considering the individual differences between the plant of the THTR and the module allows the conclusion that the steam losses may be higher than assumed by the applicant. Especially from the point of view of the evidence to be produced with respect the compliance with the intended application values for the tritium discharge through the machine housing, an appropriate method to register the discharges is to be established during the design development.

Problems concerning the radiological effects of higher steam leaks do not arise, since the intended application value for the tritium discharge through the roof of the machine housing also covers the case when the total leaks from the secondary circuit occur only as steam leaks.

3.4.1.3 Emission of Solid Radioactive Materials

During the normal operational mode, 360 fuel elements are exfiltrated per day. The partially combusted fuel elements are filled via the filling station as loose materials into transport and storage containers. Counting devices check the fill level of the container. After filling the container and the following conditioning operation takes place the mounting of the cover system. Before exfiltration, the tightness and the surface contamination of the containers is checked; and thereafter the containers are stored in an intermediate fuel element storage within the power plant site.

The intermediate space in double cover systems is monitored as to tightness via outwardly guided line connections.

Other solid radioactive materials such as, for example, used filter materials and contaminated disassembled parts are packed in collar barrels and conditioned on the spot or transported for external conditioning. The remnants are transported back to the plant. According to the safety report, an average number of 90 barrels is necessary per year, which are stored in an intermediate storage in the reactor ancillary facilities building before being transported to an external final storage. The storage capacity amounts to 200 barrels.

No problems arise concerning the environment load as a consequence of the solid radioactive materials storage.

3.4.1.4 Emission Values of Radioactive Materials with Wastewater and Exhaust Air

In the 2-module plant, the radioactive emissions via the air are composed of the following contributions emitted through the exhaust air chimney and the roof of the machine house.

Exhaust Air Chimney

The sources for the radioactive materials discharges through the exhaust air chimney are:

- The primary cooling agent leaks,
- The emissions with the ambient air of the primary cell,
- The emissions from the storage containers for radioactively contaminated helium.

The nuclide-specific differentiation of the individual emission paths as well as the total discharge through the exhaust air chimney for the design case are indicated in Table 3.4-8.

Because of the low activity inventory in the primary cooling agent, the yearly discharge through the primary cooling agent leaks is low. The iodine and aerosol-bonded fission products are reach the environment only through this path. The design parameter for the yearly iodine discharge is $9,7 \times 10^{8}$ Bq and for the of the long-lived fission products of $5,1 \times 10^{5}$ Bq. The yearly activity discharge of aerosol fission products produced by the decay of short-lived noble gases is approximately $4,8 \times 10^{9}$ Bq. Since the half-life of these radionuclides is with the exemption of Sr 89 shorter than eight days, they can be added to the noble gases, with the consequence that only the Sr 89 with $4,7 \times 10^{4}$ Bq/a must be considered among the long-lived aerosols.

The main activity rate of the fission noble gases, tritium, and C14 is emitted through the regeneration of the helium cleaning plant from the storage containers for radioactive contaminated helium. In this way, a yearly emission of $3,2 \times 10^{13}$ Bq of fission noble gases, $1,7 \times 10^{12}$ Bq of tritium, and $9,0 \times 10^{11}$ Bq of C 14 is to be expected for the design case.

The primary cell exhaust air is responsible for the discharge of Ar 41 and also for the main contribution of the aerosol activity. For the design case, a yearly activity emission of 1,2 x 10^{13} Bq of Ar 41 and 3,2 x 10^{7} Bq of aerosols is to be expected. Co 60 was selected as a representative nuclide.

Altogether, the following discharges through the exhaust air chimney are to be expected:

Gaseous radioactive materials:

Fission noble gases :	3.2 x 10^{13} Bq/a
Ar 41:	1.2 x 10^{13} Bq/a
H 3:	1.7 x 10^{12} Bq/a
C 14:	9.0 x 10^{11} Bq/a
Aerosols:	3.2 x 10^{7} Bq/a
Iodine (elementary):	9.7 x 10^{8} Bq/a
Therefrom iodine 131:	2.1 x 10^{7} Bq/a

Roof of the Machine Housing

In the design case, the following activities are derived conducted via the roof of the machine housing as a consequence of steam leaks:

H 3:	1,3 x 10^{11} Bq/a
Aerosols:	7,5 x 10^{2} Bq/a

Wastewater

The sources for the radioactive discharges via the wastewater are the purified waters from the wastewater treatment plant as well as the tritium activity being emitted with the water leaks from the secondary circuit.

In the design case, the following discharges should be expected:

H 3 5.9×10^{11} Bq/a

Fission and activation products: 1.85×10^{9} Bq/a

A nuclide-specific differentiation of the activity discharges via the wastewater path is shown in Table 3.4-9.

Table 3.4-1: Nuclide mixture of the clean water

Nuclide	Percentage of Total Vector
Co 60	24
Sr 90	0.5
Ag 110	0.5
J 131	5.0
Cs 134	15
Cs 137	55
Total	100

Table 3.4-2: Activity inventory in the evaporator strand of the wastewater treatment

Nuclide	Wastewater Coll. Tank Group I (Bq/Mg)	Evaporation Unit (Bq)	Evaporator Concentrate Tank (Bq)
Co 60	2.5×10^{7}	2.1×10^{10}	8.9×10^{10}
Sr 90	3.0×10^{4}	2.5×10^{7}	1.9×10^{8}
Ag 110m	3.0×10^{4}	2.5×10^{7}	3.7×10^{7}
J 131	9.6×10^{5}	3.2×10^{7}	3.2×10^{7}
Cs 134	1.5×10^{6}	1.1×10^{9}	4.4×10^{9}
Cs 137	9.3×10^{6}	7.4×10^{9}	5.6×10^{10}
Total	3.7×10^{7}	3.0×10^{10}	1.5×10^{11}

Table 3.4-3: Activity inventory in the centrifuge strand of the wastewater treatment

Activity Inventories

Nuclide	Wastewater Coll. Tank Group II (Bq/Mg)	Sludge Container (Bq)	Barrel 200 l (Bq)
Co 60	2.5×10^5	4.4×10^7	7.4×10^8
Sr 90	3.0×10^2	7.4×10^1	-
Ag 110m	3.0×10^2	7.4×10^1	-
J 131	9.6×10^3	2.4×10^3	-
Cs 134	1.5×10^4	3.7×10^3	-
Cs 137	9.3×10^4	2.4×10^4	-
Total	3.7×10^5	4.4×10^7	7.4×10^8

Table 3.4-4: Yearly emission through the exhaust air chimney for the 2-module plant caused by primary cooling agent leaks in Bq

Radionuclide	Yearly Emission through the Exhaust Air Chimney in Bq
Kr 83 m	8.6×10^{9}
Kr 85 m	3.1×10^{10}
Kr 85	2.0×10^{8}
Kr 87	2.9×10^{10}
Kr 88	7.0×10^{10}
Kr 89	1.2×10^{9}
Kr 90	9.6×10^{7}
Xe 131 m	7.5×10^{8}
Xe 133 m	6.8×10^{9}
Xe 133	1.4×10^{11}
Xe 135 m	2.9×10^{9}
Xe 135	8.1×10^{10}
Xe 137	2.5×10^{9}
Xe 138	1.4×10^{10}
Xe 139	1.5×10^{8}
Total noble gases	3.9×10^{11}
J 131	2.1×10^{7}
J 132	2.2×10^{8}
J 133	1.3×10^{8}
J 134	3.8×10^{8}
J 135	2.2×10^{8}
Total iodine	9.7×10^{8}

Continuation_Table_3.4-4

Radionuclide	Yearly Discharge Through the Exhaust Air Chimney in Bq
Cs 134	1.6×10^5
Cs 137	3.4×10^5
Ag 110 m	1.2×10^4
Sr 90	1.3×10^3
Total solids	5.1×10^5
Rb 88	$2.4 \times 10^9$1
Rb 89	5.0×10^8
Rb 90	2.0×10^8
Sr 89	4.7×10^4
Cs 138	1.4×10^9
Cs 139	2.9×10^8
Ba 139	2.2×10^7
Total noble gas follower	4.8×10^9

Table 3.4-5: Yearly discharge through the exhaust air chimney caused by the discharge from the storage tank for radioactively contaminated helium for the 2-module plant in Bq

Radionuclide	Yearly Activity Discharge through the Exhaust Air Chimney in Bq via the regeneration Of the Helium Purification Plant in Bq
Kr 83 m	4.1×10^{8}
Kr 85 m	4.4×10^{10}
Kr 85	2.4×10^{11}
Kr 87	1.5×10^{8}
Kr 88	2.1×10^{10}
Kr 89	-
Kr 90	-
Xe 131 m	3.3×10^{11}
Xe 133 m	5.5×10^{11}
Xe 133	3.0×10^{13}
Xe 135 m	-
Xe 135	5.5×10^{11}
Xe 137	-
Xe 138	-
Xe 139	-
Total noble gases	3.2×10^{13}
H 3	1.7×10^{12}
C 14	9.0×10^{11}

<u>Table 3.4-6:</u> Yearly discharge of radioactive materials with the exhaust air from the primary cells of the 2-module plant in Bq

Nuclide	Activity Discharge per Year in Bq
Cr 51	2.3×10^{6}
Mn 54	8.8×10^{5}
Fe 59	5.6×10^{6}
Co 58	7.5×10^{5}
Co 60	1.4×10^{7}
Ta 182	1.7×10^{7}
Total activation products	4.1×10^{7}
Ar 41	1.2×10^{13}

<u>Table 3.4-7:</u> Yearly discharge of radioactive materials conducted through the roof of the machine housing for the 2-module plant in Bq

Nuclide		Yearly Activity Discharge per Year in Bq
H 3		1.3×10^{11}
Na 24		5.6×10^{2}
Cr 51		1.1×10^{1}
Mn 54		3.8×10^{0}
Mn 56		9.4×10^{1}
Fe 59		7.5×10^{1}
Co 60		9.4×10^{0}
Ag 110 m	≅	2.0×10^{0}
Sb 124	≅	2.0×10^{0}
Total activation products		7.5×10^{2}

Table 3.4-8: Yearly discharge of radioactive materials through the exhaust air chimney in Bq (design parameters)

	Yearly Discharge through the Exhaust Gas Chimney in Bq			
Nuclide	He Leakage	Primary cell	He Storage Container	Total
H 3	7.0×10^{10}	-	1.7×10^{12}	1.8×10^{12}
C 14	7.1×10^{8}	-	9.0×10^{11}	9.0×10^{11}
Ar 41	-	1.2×10^{13}	-	1.2×10^{13}
Kr 83 m	8.6×10^{9}	-	4.1×10^{8}	9.0×10^{9}
Kr 85 m	3.1×10^{10}	-	4.4×10^{10}	7.5×10^{10}
Kr 85	2.0×10^{8}	-	2.4×10^{11}	2.4×10^{11}
Kr 87	2.9×10^{10}	-	1.5×10^{8}	2.9×10^{10}
Kr 88	7.0×10^{10}	-	2.1×10^{10}	9.1×10^{10}
Kr 89	1.2×10^{9}	-	-	1.2×10^{9}
Kr 90	9.6×10^{7}	-	-	9.6×10^{7}
Xe 131 m	7.5×10^{8}	-	3.3×10^{11}	3.3×10^{11}
Xe 133 m	6.8×10^{9}	-	5.5×10^{11}	5.6×10^{11}
Xe 133	1.4×10^{11}	-	3.0×10^{13}	3.0×10^{13}
Xe 135 m	2.9×10^{9}	-	-	2.9×10^{9}
Xe 135	8.1×10^{10}	-	5.5×10^{11}	6.3×10^{11}
Xe 137	2.5×10^{9}	-	-	2.5×10^{9}
Xe 138	1.4×10^{10}	-	-	1.4×10^{10}
Xe 139	1.5×10^{8}	-	-	1.5×10^{8}
Total	3.9×10^{11}	-	3.2×10^{13}	3.2×10^{13}

Continuation of Table 3.4-8

Nuclide	He Leakage	Primary Cell	He Storage Container	Total
J 131	2.1×10^7	-	-	2.1×10^7
J 132	2.2×10^8	-	-	2.2×10^8
J 133	1.3×10^8	-	-	1.3×10^8
J 134	3.8×10^8	-	-	3.8×10^8
J 135	2.2×10^8	-	-	2.2×10^8
Total	9.7×10^8	-	-	9.7×10^8
Cs 134	1.6×10^5	-	-	1.6×10^5
Cs 137	3.4×10^5	-	-	3.4×10^5
Ag 110 m	1.2×10^4	-	-	1.2×10^4
Sr 90	1.3×10^3	-	-	1.3×10^3
Total	5.1×10^5	-	-	5.1×10^5
Cr 51	-	2.3×10^6	-	2.3×10^6
Mn 54	-	8.8×10^5	-	8.8×10^5
Fe 59	-	5.6×10^6	-	5.6×10^6
Co 58	-	7.5×10^5	-	7.5×10^5
Co 60	-	1.7×10^7	-	1.7×10^7
Ta 182	-	1.4×10^7	-	1.4×10^7
Total	-	3.2×10^7	-	3.2×10^7
Rb 88	2.4×10^9	-	-	2.4×10^9
Rb 89	5.0×10^8	-	-	5.0×10^8
Rb 90	2.0×10^8	-	-	2.0×10^8
Sr 89	4.7×10^4	-	-	4.7×10^4
Cs 138	1.4×10^9	-	-	1.4×10^9
Cs 139	2.9×10^8	-	-	2.9×10^8
Ba 139	2.2×10^7	-	-	2.2×10^7
Total	4.8×10^9	-	-	4.8×10^9

Table 3.4-9: Yearly discharge of radioactive materials with the wastewater (design parameter)

Nuclide	Yearly Discharge Via the Wastewater Path in Bq
H3	5.9×10^{11}
Co 60	4.4×10^{8}
Sr 90	9.3×10^{6}
Ag 110 m	9.3×10^{6}
J 131	9.3×10^{7}
Cs 134	2.8×10^{8}
Cs 137	1.0×10^{9}
Total	1.85×10^{9}

Alpha emitter (Pu 239) 1.0×10^{6}

3.4.2 Radiation Exposure into the Environment

The by far largest part of the radioactive materials present in the HTR module power plant is retained in the plant, of which the main part remains in the area where they were created within the fuel particles. However, it is inevitable that small amounts of the radioactive materials penetrate into the environment when the plant is operated. The applicant cites the maximum amounts for the emission of radioactive materials, which we had assessed in Chapter 3.4.1.

The result of the emissions of radioactive materials is a radiation exposure in the environment. The radiation exposure of humans, as a result of discharge of radioactive materials with air or water must not exceed the dose limitations established in § 45 of the Radiation Protection Regulation /L 2/. This radiation exposure is to be calculated for the most adverse effect points, while taking into consideration all of the relevant load paths, including the food chains. In § 45 of the Radiation Protection Regulation /L 13/, the calculating models and parameters for calculating the radiation exposure are cited.

According to definition of the Guideline with respect to § 45 of the Radiation Protection Regulation, the radiation exposure should be understood, in the sense of § 45 of the Radiation Protection Regulation, first as the external exposure, that is, the exposure affecting humans from the outside, and second the inside radiation exposure of a member of the critical population group. A member of the critical population groups is a person, which is exposed via one or several exposure paths to the maximum radiation exposure at the most adverse effect point. Here, a realistic style of life applicable to entire parts of population should be the basis of examination. Extreme life and consumer habits should not be considered. The most adverse effect point corresponds to the point in the environment of the HTR module power plant, at which based on the distribution of the emitted radionuclides in the environment media, the highest radiation exposure is to be expected. Additionally, it is assumed that the food comes from the point with the highest food contamination in this area.

We have calculated the radiation exposure caused by the discharge of the radioactive materials by the wastewaters and the exhaust air from the HTR module power plant similarly as in the procedure of the applicant by using the calculation models and data records of the Guideline of § 45 of the Radiation Protection Regulation. In cases when no data are given in the Guideline because, for example, conditions specific to certain locations are concerned, such as, for example, meteorological data records, we have followed the assumptions of the applicant and have used the data given in the safety report. We have always checked whether these data are realistic and can be considered representative for a possible location.

However, in case of a concrete location selection, the radiation exposure calculations are to be carried out again considering all the specific features of the location, as well as other special paths, if any.

In the last section of this chapter we present the results arising from the radiation exposure calculations according to the dose model of the publications ICRP 26 /L 22/ and ICRP 30 /L 23/ with data records of the Bundesgesundsheitsamt (Federal Board of Health) /L 125, L 126/. We applied the same nuclide spectra as with the calculations according to the Guideline of § 45 of the Radiation Protection Regulation. The calculations differ only in the other dose factors.

Mathematical Models and Parameters for the Wastewater Path

In the safety report /UI 1/ and in the additional documentation /U 3.3-1/, the applicant states that 5 000 m^3/a wastewaters arise in the HTR module power plant of the control area. After the purification of the wastewater system, the wastewater released for emission contains only a mean activity concentration of 3.7×10^5 Bq/m^3. The wastewater from other areas of the HTR module power plant contains in general lower activity concentrations. To cover other operational procedures, the applicant calculates activity emissions with the wastewater that we assessed in Chapter 3.4.1.

The applicant assumes that the wastewater is entirely mixed in a receiving water with a mean flow rate MQ = 30 m^3/s. The flow rate of the receiving water is assumed to be constant the whole year.

Receiving waters with a minimum flow rate MQ = 30 m^3/s can be found in sufficient numbers (see Chapter 1). After selecting a concrete location, the distribution rate of the available receiving waters are to be examined. This is applied especially for

- The mean MQ flow rate per year,
- The mean drainage in the summer half year,
- The mixing conditions,
- The radiological previous load.

The applicant did not consider the radiological previous load yet. This cannot be done without knowing the concrete location.

The radiation exposures by the wastewater path calculated in the following text concern therefore only the emissions with the wastewater from the HTR module power plant.

Mathematical Models and Parameters for the Exhaust Air Path

Based on the modeled meteorological data concerning wind direction, wind speed, and precipitation, the applicant quantified the long-term expansion factors. The measuring points of the maximum radiation exposure with an effective emission height of 50 m were determined using the long-term expansion factors. For the calculation of the radiation exposure, the applicant applied the following long-term expansion factors /U 3.3-1/:

- For the exposure path of inhalation, beta submersion and gamma ground radiation Chi = 6,5 x 10^{7} s/m^3,

- For the exposure path of gamma submersion Chi = 6,0 x 10^{-3} s/m^3,

- For the exposure path of gamma ingestion Chi = 1,7 x 10^{-6} s/m^3.

For the tritium emissions through the roof of the machine housing, the applicant assumed an effective emission height of 18 m. For this emission path, the applicant calculates the following long-term expansion factors /U 3.3-1/:

- For the exposure path of inhalation Chi = 4,1 x 10^{-5} s/m^3,

- For the exposure path of ingestion Chi = 9,7 x 10^{-6} s/m^3.

Another measuring point was considered than with the other exposure paths for the ingestion.

The applicant took all other parameters necessary for determining the radiation exposure from the Guideline of § 45 of the Radiation Protection Regulation.

We examined the data of the applicant and the setting of the long-term expansion factors as we have performed our own calculations of the long-term expansion factors on the basis of the weather statistics characteristic for Niedersachsen. Using these statistics and the effective emission height of 50 m or 18 m, which was applied by the applicant, we found in all the paths maximum values of the long-term expansion factors that were approximately 20 % lower than those of the applicant. It is our opinion that the values used by the applicant could be applied for modeled examinations as they cover conditions for many locations.

In Chapter 3.4.1.2.2 , we also detected small emission amounts of materials in aerosol form through the roof of the machine housing. The activity emissions of these materials and the resulting radiation exposure are so small in comparison with the tritium emission through the roof of the machine housing, as well as with the emissions through the exhaust air chimney, that they contribute only in a negligible way to the total radiation exposure through the exhaust air chimney. Therefore, the data concerning the long-term expansion factors for the exposure paths of beta submersion, gamma ground radiation ,and gamma submersion are not necessary for the emissions of materials in aerosol form.

After selecting a concrete location, the data specific for this location are to be collected so that the calculation of the radiation exposure can be performed.

Radiation Exposure by Emissions with the Wastewater

In the safety report and in the additional documentation /U 3.3-1/, the applicant calculated the radiation exposures for adults and small children based on the

the wastewater discharges equal to the application values from the HTR 2-module power plant by using the models of the Guideline of § 45 of the Radiation Protection Regulation. All the exposure paths cited in the Guideline were considered taking into consideration the conservative assumptions contained there. The whole body doses determined by the applicant amount to 6.3 μSv/a for adults and to 2.2 μSv/a for small children. The highest organ dose is both for adults and small children the liver dose. It amounts to 12 μSv/a for the adults and to 4.1 μSv/a for small children /U 1, U 3.3-1/.

We have examined these calculations by performing our own radiation exposure calculations based on the Guideline of § 45 of the Radiation Protection Regulation. Only for the transfer factors did we use the data from the breakdown calculation basics /L 14/. We calculate the radiation exposure assuming a complete intermixing of the wastewater discharge with the discharge of the receiving water having a constant level of 30 m^3/s for all exposure paths throughout the year. We did not consider the radiological previous load; so that the radiation exposures concern only the radioactive discharges with the wastewater from the HTR module power plant.

For the exposure paths "sediments exposure," we assumed a shore factor of 1.0, which is valid for flowing water under the tide limit. Herewith, the dose values are also representative for the northern shore regions of Germany and cover according to the conservative approach possible locations at receiving waters over the tide limit.

For the calculation of the radiation exposure, we used as a basis a nuclide spectrum for the emission of radioactive materials with the wastewater, which we determined. Chapter 3.4.1 contains our detailed opinion on this issue.

The radiation exposures via the wastewater path calculated by us are listed in Table 3.4-10. We calculate a whole body dose of 2.6 μSv/a for adults and of 0.1 μSv/a for small children and a liver dose as the highest organ dose of 3.7 μSv/a for adults and 0.5 μSv/a for small children.

All organ and whole body doses for adults are caused by more than 60 % by the nuclide Cs 137, while for children, the contributions to the radiation exposure according to the organ or body part examined are caused by different nuclides with different weights.

The calculated radiation exposures, because of the lower activity emissions determined by us as design parameters (see Table 3.4-9), lie lower than those determined by the applicant as basic values for the application.

An additional difference is that the only path we examined for small children was the ingestion path "milk consumption," while the applicant has also examined the radiation exposure by outside radiation, swimming, boating, and sediment exposure. We do not think that the consideration of these radiation exposures in small children is necessary since for small children these long external exposure times do not occur.

The determined radiation exposures, with the whole body dose approximately at a factor of 100 and with the organ doses at more than the factor 200, lie under the limit values of § 45 of the Radiation Protection Regulation. The limit values of § 45 of the Radiation Protection Regulation are expected to be clearly exceeded if the receiving water is under slightly different conditions or if, at a concrete

Table 3.4-10: Radiation exposure by discharge of radioactive materials with the wastewater – doses per year in μSv/a (1E-06 Sv)

Adults

Path	**Bones**	**Liver**	**Whole Body**	**Thyroid**	**Kidney**	**Lung**	**Stomach/Intestine**	**Skin**
Ingestion	1.36E+00	2.05E+00	8.71E+01	1.04E+00	7.19E-01	2.54E-01	8.87E-01	
External radiation	1.68E+00	1.68E+00	1.68E+00	1.68E+00	1.68E+00	1.68E+00	1.68E+00	1.68E+00
Total	3.05E+00	3.74E+00	2.56E+00	2.72E+00	2.40E+00	1.94E+00	2.57E+00	1.68E+00

For **small children** only the ingestion of milk is considered:

Path	**Bones**	**Liver**	**Whole Body**	**Thyroid**	**Kidney**	**Lung**	**Stomach/Intestine**	**Skin**
Ingestion	4.02E-01	4.55E-01	8.54E-02	3.39E-01	6.51E-02	9.30E-02	4.62E-02	
External radiation	does not apply in wastewater path							

location, a radiological previous load is considered.. At a concrete location, the results of the radiation exposure calculation for the wastewater path must be checked with respect to all the specific conditions of this location.

Radiation Exposure by Emission with the Exhaust Air

In the safety report and the additional documentation /U 3.3-1/, the applicant presented the calculation for adults and small children based on discharges equal to the application values through the 60 m high exhaust air chimney of the HTR 2-module power plant by using the models of the Guideline of § 45 of the Radiation Protection Regulation. Because of the influences of other buildings, he expects an effective emission height of the exhaust air of 50 m. In the calculations have been taken into consideration all the exposure paths contained in the Guideline /L 13/ with the conservative assumptions contained therein.

The whole body doses determined by the applicant amount to 11,6 μSv/a for adults and 14,6 μSv/a for small children. The highest determined organ dose for adults is the bone dose with 35,6 μSv/a, for small children it is the thyroid gland dose with 19,5 μSv/a /U 1, U 3.3-1/.

We have examined the calculations of the applicant using our own radiation exposure calculations with the model of the Guideline of § 45 of the Radiation Protection Regulation and using the long-term expansion factors cited by the applicant. Otherwise, we have calculated the radiation exposure according to the parameter set of the Guideline of § 45 of the Radiation Protection Regulation, since at this time, no location-specific or special paths are available. We have used the data from the breakdown calculation documents /L 14/ merely for the transfer factors.

We have considered the location-specific characteristics only insofar as we, similar to the applicant, have based our examination on the assumption of the effective emission height of the exhaust air of 50 m, taking into consideration the influence of the building. As measuring points for the different exposure paths, we have applied the same distances from the chimney as the applicant. We consider these values for the most adverse influence points as sufficiently realistic enough in comparison with the existing power plants.

For the calculations of the radiation exposure, we started from the determined nuclide spectrum for the discharge of radioactive materials with the exhaust air. Chapter 3.4.1 contains our detailed opinion on this issue.

The radiation exposures through the exhaust air path, which we calculated, can be found in Table 3.4-11. We calculate an whole body dose of 11,1 μSv/a for adults and of 19,7 μSv/a for small children. The highest organ dose for adults found is the bone dose with 43,5 μSv/a, for small children it is the thyroid dose with 24,3 μSv/a /U 1, U 3.3-1/.

In adults, the radiation exposure for the whole body and all organs with the exception of the skin is caused in the nuclide spectrum used as more than 50 % by the nuclide Ar 41; additionally, the nuclides Ar 41 and Xe 133 deliver a substantial contribution to the dose. In small children, the contribution of C 134 with respect to the total dose for the whole body and the most organs is more than 70 %, with the does for the thyroid being about 60 %, a 14 % rate caused by J 131 is added. The rest of the dose is caused mostly by the nuclides Ar 41 and Xe 133. These two noble gas nuclides cause in adults as well as in small children also approximately 95 % of the skin dose.

The radiation exposures caused by the exhaust air path determined by us lie at about 20 % for adults and at a maximum of 35 % in small children above the values determined by the applicant but far below the limit values of § 45 of the Radiation Protection Regulation.

The differences in the calculations are caused only in small children by the exposure path of the milk consumption as well as in the adults by the exposure paths of milk and meat consumption. All other dose values of the applicant can be confirmed. This may be caused by the different way of applying the transfer factors.

Since the radiation exposures lie considerably below the limit values of § 45 of the Radiation Protection Regulation, the different results of the calculations for the assessment are not important. It is our opinion that the limit values of § 45 of the Radiation Protection Regulation have clearly not been exceeded also in the selection of the concrete location with the presented concept. Aside from this, the radiation exposure calculation should be tested with respect to the conditions specific for this location.

Table 3.4-11: Radiation exposure caused by the emission of radioactive materials with the exhaust air – doses per year in µSv/a (1E-06 Sv)

Adults

Path	Bones	Liver	Whole Body	Thyroid	Kidney	Lung	Stomach/ Intestine.	Skin
Ingestion	3.91E+01	6.79E+00	6.79E+00	7.55E+00	6.78E+00	6.79E+00	6.88E+00	
Inhalation	1.28E-01	2.95E-02	2.94E-02	3.45E-02	2.94E-02	3.01E-02	2.98E-02	
Beta subm.								2.89E-01
Gamma subm.	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00
Gamma ground	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02
Total	4.35E+01	1.11E+01	1.11E+01	1.19E+01	1.11E+01	1.11E+01	1.12E+01	4.61E+00

Small Children

Path	Bones	Liver	Whole Body	Thyroid	Kidney	Lung	Stomach / Intestine.	Skin
Ingestion	1.51E+01	1.53E+01	1.53E+01	1.99E+h01	1.53E+01	1.53E+01	1.53E+01	
Inhalation	3.45E-02	4.04E-02	4.03E-02	4.92E-02	4.04E-02	4.11E-02	4.04E-02	
Beta subm.								2.89E-01

Gamma sub.	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00
Gamma ground	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02
Total	1.94E+01	1.97E+01	1.97E+01	2.43E+01	1.97E+01	1.97E+01	1.97E+01	4.61E+00

Radiation Exposure by Emissions Through the Roof of the Machine Housing

In the safety report, the applicant states that besides the discharge of radioactive materials through the exhaust air chimney, H 3 is also discharged through the roof of the machine housing because of steam leaks from the secondary circuit. The H 3 is, according to the information of the applicant, the only nuclide, which is drawn away through the roof of the machine housing into the environment.
The discharges take place through the ventilators 3 m above the 25 m high roof of the machine housing at a height of 28 m. The emissions height caused by the influence of the building is 18 m.

The results of the radiation exposure calculations of the applicant caused by the discharge of the H 3 through the roof of the machine housing are contained in the values cited the applicant in the previous chapter.

We have examined the calculations of the applicant by carrying out own radiation exposure calculations for H 3 with the models of the Guideline of § 45 of the Radiation Protection Regulation and by using the long-term expansion factors mentioned by the applicant and also the tritium discharges through the roof of the machine housing cited by us in Chapter 3.4.1. These tritium discharges are lower that those cited by the applicant as values for the application. When calculating the radiation exposure caused by the H 3, we took into consideration the exposure paths examined in the Guideline of § 45 of the Radiation Protection Regulation. The measuring points amount to 100 m for the exposure path of inhalation and 270 m for the exposure path of ingestion.

We come to the result that by the discharge of H 3 through the machine house roofs a maximum radiation exposure of 0,2 μSv/a for all organs and the whole body is being caused. Herewith, the dose caused by the removal of radioactive materials with the exhaust air is being increased by approximately 2 %. The limit values of § 45 of the Radiation Protection Directive are far from being reached.

Radiation exposure caused by direct radiation from the plant

In the Safety Report the applicant did not give any information on the radiation exposure caused by direct radiation from the plant.

The radiation exposure caused by direct radiation is being determined by the local dose rate at the fence. We have stated our stand on this issue in Chapter 3.1.2. We come to the result that the radiation exposure caused by direct radiation from the plant can be neglected.

Total radiation exposure

In the Tables 3.4.-12 and 3.4-13, we collected the total radiation exposure caused by wastewaters and exhaust air path we found and compared it with the limit values of the § 45 of the Radiation Protection Directive. With the thyroid gland, the total dose over both paths caused by the ingestion dose is to be considered. The nuclide spectra stated in the Chapter 3.4.1 were the basis.

The results show that the limit values of § 45 of the Radiation Protection Regulation are far from being reached. However, we must point out that, according to the Radiation Protection Regulation, also the previous load caused by other nuclear facilities or other users of radioactive materials should be taken into consideration. Since at the time being, no concrete location for the HTR module power plant has been selected yet, these influence factors are not contained in Tables 3.4-12 and –13. The tables reflect only the radiation exposures caused by the HTR module power plant. In comparison with the radiation exposure caused by the discharge of radioactive materials with wastewater and exhaust air, the radiation exposure coming from the plant plays only a negligible role. Thus, the limit value according to § 45 of the Radiation Protection Regulation is far from being reached.

The selection of a concrete location can cause a change in the obtained dose values because the specific conditions of the location, for example, the drainage conditions of the receiving water, the meteorological, the terrain, and the construction conditions in the surrounding area of the facility, or the particular eating customs of the population, can also influence the radiation exposure calculations. After selecting a location, the calculation of the radiation exposure must be tested by the ordinary operation of the HTR module power plant.

Since he dose values determined until now are still clearly below the limit values of § 45 of the Radiation Protection Regulation, the compliance with the limit values will also be possible after selecting a location.

Table 3.4-12:Overall radiation exposure caused by the discharge of radioactive materials from the

Adults	Bones	Liver	Whole B.	Thyroid	Kidney	Lung	Stom./Int.	Skin
Wastewater	3.1	3.7	2.6	2.7	2.4	1.9	2.6	1.7
Exhaust air (chimney and machine housing)	43.5	11.3	11.3	12.1	11.3	11.3	11.4	4.6
Direct radiation	-	-	-	-	-	-	-	-
Limit value acc. R.P. Regulation	1800	900	300	900	900	900	900	1800

*) In the thyroid dose. the total of the ingestion dose values by the wastewater and exhaust air path should be compared with the limit value of the Radiation Protection Regulation.

Table 3.4-13: Overall radiation exposure caused through the discharge of radioactive materials from the HTR module power plant in μSv/a

Small children	Bones	Liver	Whole b.	Thyroid	Kidney	Lung	Stom./Int.	Skin
Wastewater	0.4	0.5	0.1	0.3	0.1	0.1	0.1	-
Exhaust air (chimney and machine housing)	19.4	19.8	19.8	24.4	19.8	19.8	19.8	4.6
Direct radiation	-	-	-	-	-	-	-	-
Limit value acc. R.P. Regulation	1800	900	300	900	900	900	900	1800

*) In the thyroid gland dose, the total of the ingestion dose values through the wastewater and exhaust air path should be compared with the limit value of the Radiation Protection Regulation.

Radiation Exposure Caused by a Discharge of Radioactive Materials Equal to the Values of the Application

In the safety report, the applicant cited application values for the discharge of radioactive materials with the wastewater and the exhaust air. Additionally, the applicant cited nuclide spectra in the safety report, which were determined on the basis of the values of the application.

In the safety report, the applicant carried out radiation exposure calculations on the basis of the values of the application. The applicant arrives at the result that the determined radiation exposures are under the limit values of § 45 of the Radiation Protection Regulation.

In the wastewater path, the values of the application for H 3 are lie over the values determined by us in Chapter 3.4.1 by the factor 6,5 and for the remaining nuclide mix by the factor 5. In the discharges through the exhaust air chimney, the values of the application for H 3 and the fission noble gases are higher by up to the factor 2 over the values that we had determined; for Ar 41, C 14, the iodine isotopes, and the long-lived aerosols, they are approximately on the same level. In the tritium discharges through the roof of the machine housing, the value of the application is higher approximately by the factor 6,5 than the values we determined.

We examined the radiation exposure calculations of the applicant. Because of the higher released source energy level when defining the values of the application, the radiation exposures are a little higher than we mentioned in Tables 3.4-10 to 3.4-13. However, the differences are much lower then one order or magnitude in all exposure path/paths.

On the basis of our radiation exposure calculations, we confirm the statements of the applicant that the radiation exposure caused by the discharge of radioactive materials at the level of the values if the application will be well below the limit values of § 45 of the Radiation Protection Regulation, both in the wastewater path and in the exhaust air path.

Calculation of the Radiation Exposure with Dose Factors According to ICRP 30/BGA

In its new dose limiting concept published in 1977 in the publication ICRP 26 /L 22/, the International Radiation Protection Commission introduced the term "effective equivalent dose" as the dose parameter for the entire radiation risk. To calculate the effective equivalent dose, the weighted equivalent doses of all the relevant organs or tissues are added; wherein the weighted factors take into consideration the relative radiation sensitivity of the individual tissues and also the severity of the possible radiation damage.

In connection with the introduction of the concept of the effective equivalent dose, the International Radiation Protection Commission presented in 1978 in its publication ICRP 30 /L 23/, improved calculation models for calculating the organ doses for the organs and tissues, which essentially contribute to the effective dose. The basis for these calculation models are improved metabolism models for the respiratory tract and the digestive tract as well as a dosimeter model for the bones.

The concept of the effective equivalent dose and the dose model of the recommendation in the IRCP 26/30 have been for some years the basis of the radiation exposure protection legislation according to the guidelines of the Council of the European Union. The Federal Republic of Germany is bound to implement the guidelines of the council in the national legislation. The respective new version of the Radiation Protection Regulation will be in force since November 1989. Therefore, we have determined the potential radiation exposure for the HTR module power plant also with the dose factors for inhalation and ingestion on the basis of the ICRP 26/30. In our calculation, we applied the dose factors calculated by the German Federal Board of Health based on the models ICRP 26/30 /L 125, L 126/. The dose values in Tables 3.4-14 and 3.4-15 differ only in the dose factors used in the calculations from the values shown in Tables 3.4-10 and 3.4-11. All other calculation values are identical. To make comparisons possible, we also applied the nuclide spectra cited in Chapter 3.4.1 when carrying out the radiation exposure calculations.

A comparison of the radiation exposures calculated with the different dose factors shows that there are only slight differences when comparing comparable organs. With comparable doses are detected both a little higher and also a little lower values, which can be assumed to be caused by the different assessment of some nuclides in the models of ICRP 26/30. From the individual organ doses is calculated the effective equivalent dose with the respective discrimination according to the ICRP 26/30 models.

All the values for the individual organ doses as well as for the effective equivalent dose determined with the does factors according to ICRP 30/BGA lie clearly below the limit values of the new version of the Radiation Protection Regulation. This applies also in the case in which, instead of the nuclide spectrum, which we used as a basis in Chapter 3.4.1, the values of the application from the Safety Report are being used as source energy levels for the radiation exposure calculations.

Table 3.4-14: Radiation exposure caused by the discharge of radioactive materials with wastewater according to ICRP 30/BGA – doses per year in µSv/a (1E-06 Sv)

Adults

Path	Eff. Eq.	Bone Marrow	Bone Surface	Thyroid	Lung	Liver	Colon	Skin
Ingestion	9.93E-01	9.32E-01	9.57E-01	1.05E+00	8.89E-01	1.01E+00	1.03E+00	
Ext. rad.	1.68E+00	1.68E+00	1.68E+00	1.68E+00	1.68E+00	1.68E+00	1.68E+00	1.68E+00
Total	2.68E+00	2.62E+00	2.64E+00	2.74E+00	2.57E+00	2.69E+00	2.72E+00	1.68E+00

For **small children** is only taken into consideration the ingestion of milk:

Path	Eff. Eq.	Bone Marrow	Bone Surface.	Thyroid	Lung	Liver	Colon	Skin
Ingestion	5.68E-02	5.19E-02	5.48E-02	2.70E-01	4.54E-02	4.74E-02	5.22E-02	
External radiation	does not apply for wastewater path							

Table 3.4-15: Radiation exposure caused by the discharge of radioactive materials with the exhaust air according to ICRP 30/BGA – doses per year in µSv/a (1E-06 Sv)

Adults

Path	Eff. Eq.	Bone Marrow	Bone Surface	Thyroid	Lung	Liver	Colon	Skin
Ingestion	2.18E-01	2.18E-01	2.18E+01	2.24E+01	2.18E+01	2.18E+01	2.19E+01	
Inhalation	8.20E-02	8.18E-02	8.18E-02	8.38E-02	8.29E-02	8.18E-02	8.18E-02	

Beta Subm.								2.89E-01
Gamma Subm.	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00
Gamma Ground	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02
Total	2.63E+01	2.62E+01	2.62E+01	2.78E+01	2.62E+01	2.62E+01	2.63E+01	4.61E+00

Small Children

Path	Eff. Eq.	Bone Marrow	Bone Surface	Thyroid	Lung	Liver	Colon	Skin
Ingestion	4.89E-01	4.88E+01	4.88E+01	5.24E+01	4.88E+01	4.88E+01	4.88E+01	
Inhalation	1.52E-01	1.51E-01	1.51E-01	1.56E-01	1.61E-01	1.51E-01	1.51E-01	
Beta Subm.								2.89E-01
Gamma Subm.	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00	4.31E+00
Gamma Ground	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02	1.44E-02
Total	5.34E+01	5.33E+01	5.33E+01	5.69E+01	5.33E+01	5.33E+01	5.33E+01	4.61E+00

3.5 Radiation and Activity Monitoring

3.5.1 Monitoring System

The monitoring system includes, in accordance with the safety report, the measurement of activities and concentrations of radioactive materials in the circuits during the ordinary operation and when breakdowns occur. Should limit values be exceeded, reports are sent to the control room so that measures can be taken to control the emissions or eliminate the faults.

The following systems with fixedly installed continuously measuring devices are monitored individually

- The nuclear intermediate cooling system and accessory cooling water system,
- The safeguarded intermediate cooling system and accessory cooling water system,
- The accessory steam condensate system, and
- The helium purification plant, including the ancillary helium systems.

From following circuits, samples are taken at random and examined in laboratories with respect to nuclides:

- The water steam circuit,
- The process steam system,
- The nuclear intermediate cooling system and ancillary cooling water system, and
- The ancillary helium systems.

Our assessment was performed according to the Safety Criteria for Nuclear Power Plants, Chapter 10, Radiation Protection Monitoring /L 6/. In addition, the RSK Guidelines for pressurized water reactors, Chapter 10, Radiation Protection Monitoring /L 10/ and the KTA Regulation 1504 /L 64/ were utilized.

With the measurement and sample taking devices for system monitoring provided in the safety report are fulfilled the requirements of the Safety Criteria for Nuclear Power Plants and the RSK Guidelines, Chapter 10, concerning the task and scope of the monitoring.

In contrast with the KTA Regulation 1504 and the RSK Guidelines, the water steam system is monitored not by continuous but by random sampling. In the documentation /U 3.5-1/ are stated the causes for the random monitoring system. We examined the data and we have no reservations against the intended monitoring of this circuit. Depending on the intended use for the process steam, a decision has to be made in the design development as to whether the process steam monitoring has to be carried out as planned via sampling and at random.

3.5.2 Room Monitoring

The local dose rate monitoring should serve to protect the operation personnel against radioactive external radiation. For this purpose, the local dose rate should be measured in the rooms or room groups in which restrictions concerning the time spent in them are imposed because of the expected local dose rate but which the employees enter as a matter of routine. Another task of these measuring points is to warn the personnel when the limit values have being exceeded.

According to the safety report, measurements are performed in the above-mentioned rooms by using fixedly installed devices and portable devices in rooms with very different local radiation levels. Additionally, two breakdown-proof measuring devices monitor the reactor hall.

The requirements mentioned in the KTA Regulation 1501 /L 62/, in the RSK Guidelines /L 1/, and in the BMI Safety Criteria /L 6/ can be fulfilled with respect to their objective and scope by the measuring devices cited in the safety report.

We have no reservations with respect the concept of the local dose rate monitoring.

Ambient Air Monitoring

According to the safety report and the technical documentation /U 3.5-2/, it is planned to take samples of the ambient air of the ventilation/exhaust air strands of the HTR module, the reactor hall, including the supply section, the helium section in the ancillary facilities building, and the machine housing, and to then feed these to measuring device to monitor for radioactive materials. With the exception of the machine housing, all the air samples are tested for noble gases, aerosols, and tritium by using fixedly installed, continuously measuring devices. The machine housing exhaust air is checked for tritium.

The scope of the measuring devices for monitoring the ambient air for radioactive materials presented in the safety report and the technical documentation /U 3.5-2/ takes into consideration the assignment of the ventilation system specific for the facility. By using the provided monitoring devices are fulfilled the requirements of the safety criteria /L 6/. In addition, the measuring tasks described in the KTA Regulation Design Proposal 1502.2 /L 6/ can be carried out. We have no reservations against the projected ambient air monitoring.

3.5.3 Activity Discharge Monitoring

Exhaust Air Monitoring

The monitoring of the exhaust air includes, in accordance with the safety report

- Radioactive noble gases,
- Radioactive aerosols,
- Radioactive iodine,
- Tritium and
- Carbon 14.

The radioactive noble gases and aerosols are monitored with fixedly installed, continuously measuring facilities.

The balancing of the above-mentioned materials is performed in the noble gases with a nuclide-specific measuring device, in the aerosols and iodine by balancing the collection filters. Tritium and carbon 14 are drawn off as samples from the exhaust air stream and are balanced.

In principle, the measuring devices for monitoring the activity discharge provided in the safety report comply with the requirements resulting from the KTA Regulation 1503.1 /L 63/. However, we want to warn as early as at this stage, that the balancing as to radioactive strontium and alpha emitter has to be included into the balancing program for the chimney exhaust air.

Monitoring of the Radioactive Wastewater

The monitoring of the radioactive wastewater includes, in accordance with the safety report

- The machine housing wastewater,
- The wastewater from the condensate purifying plant,
- The water form the wastewater control tank, and
- The overall wastewater.

The monitoring of the machine housing wastewater is carried out indirectly by taking samples from the water steam circuit and evaluating the discharged water amount. The wastewater from the condensate purifying plant is being collected in a container and tested for activity before drainage. Continuously measuring devices are provided for the monitoring of the discharges from the wastewater control tank and the overall wastewater quantity.

With the radioactive wastewater monitoring concept presented in the safety report are met the requirements of the Safety Criteria for Nuclear Power Plants /L 6/. In contrast with the KTA Regulation 1504, the machine housing wastewater is monitored at random by sampling. In the documentation /U 3.5-1/ are provided the reasons for the random monitoring. We examined the data and we have no reservations with respect to the intended procedure.

3.5.4 Breakdown Monitoring

With the measuring devices for monitoring breakdowns should be possible an assessment of the state of the plant before, during, and after the breakdown in the case of breakdowns and events that can result in increased discharge of radioactive materials into the environment. The measuring devices are divided according to their tasks in correspondence with the KTA Regulation 3502 /L 65/ into breakdown indication and breakdown recording devices. The breakdown indication consists of

- The overview breakdown indicator,
- The detailed breakdown indicator, and
- The wide-range indicator.

According to the safety report, two breakdown-proof measuring devices should be installed for the monitoring of the local dose rate in the reactor hall to perform these tasks. The discharge of radioactive materials with the exhaust air should be monitored at a breakdown-proof noble gas measuring point. In the case of a pressure relief breakdown, the released primary cooling agent is discharged directly through the relief valves in the reactor and the ancillary reactor building without filtration. The inlet points are situated above the sampling point for the measuring devices for the exhaust air monitoring. During this accident, the discharge is monitored by an iodine sample collecting device. The filter cartridge is tested in the laboratory. The noble gas fraction is calculated from the continuous noble gas monitoring of the primary cooling agent.

The measuring facilities provided and mentioned in the safety report can comply with the requirements of the overview breakdown indicator according to the KTA Regulation 3502 /L65/.

With the provided measuring devices is possible a detailed breakdown indication in accordance with the KTA Regulation 3502 /L 65/ with reference to their objective and scope.

In principle, the above-mentioned measuring devices can also meet the requirements concerning the wide-range indicator. However, we want to warn as early as at this stage, that the measuring range of the intended devices for measuring the ion dose rate is not sufficient and will have to be correspondingly adapted

3.5.5 Personnel Monitoring

With the personnel monitoring system can be monitored the body doses and contaminations of persons who enter the control area. In addition to the measurement of the personal dose using dosimeters, all persons should be tested for contamination when leaving the control area. According to the safety report are provided hand-foot-clothing monitors. Furthermore, portable contamination monitors should be used within the control area.

The requirements concerning the monitoring of the physical radiation protection contained in the Radiation Protection Regulation /L 2/ are met with the use of the concept for monitoring persons described in the safety report. We already referred to the necessity to providing dosimeters in accordance to the state of the art in Chapter 3.3.2. We have no reservations with respect to the concept of monitoring persons.

3.5.6 Environmental Monitoring

The environmental monitoring completes the monitoring of the activity discharge and plans to monitor the sections "Air and Precipitation," "Ground and Vegetation," "Terrestrial Nutritional Chains," "Waters and Groundwater," and "Aquatic Nutritional Chains." In accordance with the safety report, the monitoring of the environment is classified into the following three monitoring tasks:

- Auditing before putting the facility into operation,
- Monitoring the discharge of radioactive materials during ordinary operation, and
- Monitoring the emission of radioactive materials into the environment caused by breakdowns.

Using the concept for environmental monitoring presented in the safety report, the BMI requirements concerning the monitoring of emissions and immissions of nuclear reactor plants /L 16/ can be met. An assessment of the type of monitoring, the installation points, and the compliance with the requirements concerning the measuring devices to be utilized is not performed for concept assessment. These aspects are to be evaluated during the design development.

4 Operation of the Plant

4.1 Overview

According to the presentation in the safety report /U 1/, the HTR 2-module power plant is projected in the form of two identically constructed half-load plants. Each section (module) is designed for a thermal reactor performance of 200 MW and for a thermal steam generator performance of 202 MW, consequently the thermal block performance is 400 or 404 MW. Each section of the plant consists of the primary side with the nuclear steam generating system with the reactor, the primary circuit blower, the steam generator, and the systems for pressure regulation and pressure relief, as well as the assigned secondary side as water steam circuit.

In principle, each secondary side consists of a feed water container, two main feed water pumps, a turboset, a process steam tapping station with a process steam superheater, a fresh steam reducing station, a low pressure preheating section, a unit supplying de-ionized water from the plant network, a safety valve, a steam generator relief device, and a start-up and shut-down circuit.

Each one of the turbosets consists of a three housing steam turbine, an air-cooled generator with a maximum electric performance of approximately 63 MW, and a condensation plant; wherein the steam turbine consists of a high pressure turbine, a preheating (medium pressure) turbine, and a condensation turbine (low pressure turbine).

The nominal fresh steam flow rate of a steam generator amounts to 77 kg/s with a pressure of approximately 190 bar at the generator outlet and a temperature of approximately 530 ° C. This overheated fresh steam is first transported to the high-pressure turbine and a part of it arrives as exhaust steam into the medium pressure and low pressure turbine. Another part is transported as process steam through the process steam super heater to the process steam tapping station. The heating steam for the process steam superheater is also extracted from the high pressure turbine through a tapping point. The process steam leaves the tapping station with a temperature of 272 ° and a pressure of 17 bar; wherein the quantities provided for the winter operation of the tappng throughput are of 54,7 kg/s, and for the summer operation of 23,6 kg/s. With a process steam tapping of 54,7 kg/s, the generator performance is about 42 MW /U 4.1.-1/. The steam for the low pressure preheating and the auxiliary steaming of the feed water tank is extracted through tappng points of the medium pressure turbine; the exhaust steam is transported from the low pressure and medium pressure turbine into a condenser designed for a heat rate of approximately 70 MW /U 4.1-2/.

In parallel with the turboset is arranged a fresh steam pressure reducing station designed for a fresh steam flow rate of 77 kg/ is (100 %). Via this station can be diverted the whole fresh steam in the case of a partial or total breakdown of the turboset. The fresh steam can be diverted into the process steam tapping point, into the process steam superheater, into the condenser, and into the neighboring plant. Further connections on the fresh steam side between both half-load plants are at the input sides of the turbines or the fresh steam pressure reducing stations.

The exhaust steam condensed in the condenser is conducted as condensate via two main condensate pumps (on each side) and over the low pressure preheating line to a feed water collecting tank common for both plants, and then to the feed water containers, which can also be connected to each other on the outlet side. To equalize the extracted process steam, de-ionized water can be added from the plant network over a de-ionized water store tank and via additional pumps before the low pressure preheating lines or in the feed water collector. By connecting the two plants on the fresh steam and process steam side, the operation of the steam utilization systems can be partially or fully operated.

Additionally, for each of the two plants is provided a separate reactor protecting system, which switches off the respective plant in the case of a fault or breakdown, and subsequently disconnects the latter from the secondary side; the operation of the other facility can continue.

The control and regulating devices of both facilities consist of a high-level block control system for the whole plant and of lower level control systems on the primary and secondary side for each plant. Their task is to keep the process variables important for the process control within their appropriate ranges. In the block control system the required values for the thermal performance with respect to the process steam rate, to the generator performance to be achieved, and with respect to the limitations based on the highest possible performance levels of the modules are being created for the different plants. Here, the process steam throughput has priority over the current generation. In the primary and secondary side control systems of the facilities, the following regulated and actuating variables are provided:

	Control Variable	Regulating Variable
<u>Primary side:</u>	thermal steam generator perf.,	feed water flow rate,
	fresh steam temperature,	rotation speed of the primary circuit blower,
	hot gas temperature,	reflector rods,
<u>Secondary side:</u>	fresh steam pressure,	control valve high pressure turbine,
	max. fresh steam pressure,	fresh steam diversion to process steam system,
	process steam temperature,	control valve high pressure turbine tappng point,
	min. high pressure extracted steam pressure,	fresh steam diversion to high pressure tapping line,
	process steam pressure,	control valve low pressure turbine,
	max. process steam pressure 1,	safety control valves/overload low pressure and medium pressure turbine,
	max. process steam pressure 2,	steam diversion to condenser,
	min. process steam pressure,	fresh steam diversion to process steam system,
	min. feed water cont. pressure,	fresh steam diversion to medium pressure bleed. syst.,
	back pressure medium pressure turbine,	control valves medium pressure turbine,
	max. back pressure medium pressure turbine,	medium pressure turbine exhaust steam diversion to condenser,
	rotation speed turbine,	all turbine control valves.

4.2 Operating Modes

4.2.1 Start-up and Shut-down, Full-power Operation, Revision

Start-up

While starting up a module, difference is made between a cold and a warm start.

The initial state for a cold start up is a primary circuit pressure of about 38 bar with a temperature of approximately 50 ° C. The systems on the secondary side including the start-up and shut-down circuit and the other accessory and auxiliary systems are in an operational state. After connecting the primary circuit blower, the helium flow rate is brought to approximately 40 % and the feed water and fresh steam flow rate is brought to approximately 20 % as well as the secondary pressure to approximately 70 bar by using main feed water pumps with rotation speed regulation. After the reactor is made critical in this state, an increase of the hot gas temperature with a gradient of approximately 2 K/min follows. Resulting from the increase of the helium temperature and with the assistance of the primary pressure regulation the primary pressure rises slowly to its nominal value of about 60 bar. From a hot gas temperature of approximately 450 ° C, the control systems for the hot gas temperature, the fresh steam temperature, the fresh steam pressure, and the feed water flow rate are put into operation; afterward, the hot gas temperature is again increased by approximately 2 K/min. After reaching a fresh steam state of 530 ° C and 190 bar, the turboset and the other water steam circuits can be put into operation and the start-up and shut-down circuit can be shut off. The further start-up process up to the nominal performance is performed via a high level block control system and the already mentioned lower level control systems. The described start-up operation lasts approximately 6-8 hours.

After a high-speed shut-off of the reactor (RESA), a warm start of the respective plant can follow only within approximately 1 hour after the high-speed shut-off because, from this moment on, the neutron-physical intoxication influence of the xenon development becomes too high. The initial state during a warm start corresponds, with the exemption of the small heating up of the core depending on the operation time, to the state after the high-speed shut-off, that is, the primary and secondary throughputs are interrupted, while the pressures on the primary side and in the steam generator remain almost unchanged. The hot start is initiated by putting into operation of the primary circuit blower, the steam generator, and the start-up and shut-down circuit, wherein first a primary and a secondary rate are set to approximately 10 %. After the control systems for the fresh steam pressure and for the fresh steam temperature are put into operation, the reactor can again be made critical after the release of the reflector rods approximately 10 minutes after the blower start. The heating up of the core developed after the high-speed shut-off is again decreasing over the secondary side. The transition from the start-up and shut-down circuit to the other devices of the water steam circuit is carried out after the connection of the control systems for the hot gas temperature and the thermic steam generator. Using the block control system, the thermal performance with a gradient of approximately 5 %/min is brought to a performance of at least 45 %; the required values for the different regulated variables are implemented/registered according to the partial-load diagram.

Shut-down

While shutting-down a module, differences are made between a shutting-down from the full-load or partial-load operation and between a shut-down after a high-speed shut-off.

The shutting-down from the full-load operation corresponds essentially to a reversion of the procedure performed during the cold start. With the aid of the block control systems and the lower level systems of the respective module, the operation is at first changed to a lower partial-load point of about 20 %; the hot gas temperature is reduced with a gradient of approximately 2 K/min, and the helium flow rate and the steam generator performance are guided in accordance with the partial-load diagram

After reaching this operational state, the block control system and the steam generator control with a feed water flow rate of approximately 20 % are shut off and the heat discharge is taken over by start-up and shut-down circuit. With the hot gas temperature further dropping with a gradient of approximately 2 K/min, the fresh steam temperature and the fresh steam pressure are also lowered to a minimum value of 70 bar. As a further consequence of the dropping hot gas temperature, the helium throughput is increased by an intervention of the fresh steam temperature control system to a level of approximately 40 %; after this value is reached, the control systems for the hot gas temperature and the fresh steam temperature are shut off. The last phase of the shut-down operation is carried out manually: after the conversion of the start-up and shut-down circuit to the water operation and after the nuclear switch-off has been carried out, the primary pressure control system is shut off and the long-term secondary heat discharge is passed to the start-up and shut-down circuit.

The initiation of the shut-down operation within 1 hour after the high-speed shut-off corresponds at first to the procedure during a warm start, that is, after putting into operation the primary circuit blower and of the start-up and shut-down circuits, the helium throughput and the feed water throughput are set at approximately 10 %. In this state the plant cools down with a gradient of approximately 1-5 K/min; wherewith the cooling down, also the fresh steam temperature and the fresh steam pressure are decreased via their control systems until the conversion to the water operation is reached. Thereafter, the long-term secondary heat discharge is again performed via the start-up and shut-down circuit. With respect to the case of a long-term shut-down after a high-speed shut-off, see Chapter 4.2.3 (Secondary heat).

Full-power Operation

In the full-load state, on the primary side, the value of a hot gas temperature lies at about 700 ° and a cold gas temperature lies at approximately 250 °, with a helium flow rate of approximately 86 kg/s and a primary pressure of approximately 60 bar; on the secondary side, in the steam generator, the value of a fresh steam temperature lies at approximately 530 °, a feed water temperature lies at approximately 170 °, a feed water and fresh steam flow rate lies at 77 kg/s, and a fresh steam pressure lies approximately at 190 bar. The coherence between these process variables is stated in the partial-load scheme for the performance range between 20 % and 100 % in the safety report. In the partial-load range is provided an unlimited range between 50 % and 100 % and a limited range between 20 % and 50 %, depending on the xenon poisoning (see here our limitation in Chapter 2.5.5.1).

When there are performance changes, there is a differentiation between a rapid change with constant hot gas temperature and a long-term change with changing hot gas temperature. The performance change of a module is carried out with the assistance of the high level block control systems and the lower level control systems of the respective module; the required values of the most important process variables can be modified according to the partial-load diagram.

In the safety report is described how the start-up operation, the shut-down operation, and the full-load and partial-load operations should be carried out. The performance-dependant behavior of the different process variables is mentioned in the partial-load diagrams. The occurring core and components temperatures are under the respective temperatures occurring in cases of faults and breakdowns. We refer to Sections 4.2.2 (Reactor High-speed Shut-off), 4.2.3 (Secondary Heat Drainage)