The sprinkle time is estimated at approximately 60 s and the sprinkled area is assumed to be equal to 1/10 of the preheater heating surface. The steam interacts with the entire steam generator surface for almost 5 hours. For J and Sr, 50 % of the Cs reaction rates are applied. The aforementioned assumptions lead to steam generator activity remobilization of 2.1 E+09 Bq of J 131, 1.7 E+10 Bq of Cs 137, and 5.9 E+07 of Sr 90 at the preheater as leakage location.

- Mobilization by Fuel Element Corrosion

Approximately 240 Kg of fuel element graphite are enclosed with water until the closure of the safety valves after approximately 5.5 hours, according to the note (D 3.3-2). This leads to a fuel element corrosion of up to 3 weight percent. A fuel zone corrosion does not occur. The specific graphite matrix activity is determined by the silver and uranium contamination of the natural graphite. Except for the surface, the activities obtained in this way correspond to the activity profiles for Cs 137 and Ag 110 m (5.8.1-2). The aforementioned assumptions result in a mobilization by the fuel element corrosion of 2.1 E+09 Bq of J 131, 1.7 E+08 Bq of Cs 137, 1.3 E+08 Bq of Sr 90, and 3.0 E+07 Bq of Ag 110 m.

- Iodine and Noble Gas Release bya UO_2 Oxidation

In the attached documentation (D 5.8-6, D 5.8-7), the applicant presents the possible increase of the fission product release in the event of water penetration into the reactor core by UO_2 oxidation. The process described below is generally possible. The steam penetrates by diffusion into the porous graphite material and transforms partially into water gas.

The latter penetrates further by diffusion through the cover layer fissures of the coated particles as UO_2 and oxidizes the UO_2 into UO_{2+x}. This oxidation could lead to structural changes (grain boundary segregation, grain growth, phase transformation) of the UO_2 and threfore to changes in the diffusion velocity of the fission products and an increased fission product release.

In the above-mentioned documents, the applicant discusses the reaction kinetics of this process and reaches a quantitative estimate of the related release rates for Xe 133 and J 131.

The filtration efficiency degrees used by the applicant are relevant except for iodine. We do not take into consideration those which are similar to the approach in the event of a core heating disruption due to the a missing redundancy. The organic iodine compositions are estimated by us to be at 50 %. Considering this deviation from the applicant's assumptions, in Table 5.8.1.-4 results the activity release data at the chimney location resulting from a heating pipe leakage in the preheater area. This nuclide vector covers a break in a heating pipe also in the floor callott area.

In the event of leaks in the heating pipe in the overheater and the generator result higher release values of radioactive substances because of the dominance of the contribution by remobilization via the water jet by the solid matters. The values for the overheater area correspond then to those for the steam generator sector. The emissions nuclide vector for leaks in the heating pipe in the area of the overheater have been additionally determined by us (s. Table 5.8.1-5).

The high fraction of activity that is bound by adsorption in the steam generator and therefore remobilizable, related to radioactive substances, which are sedimented to the primary circuit inventory, are in our opinion the result of a conservative estimate (ifor eample, Cs 76 %). With respect to the break in a heating pipe diameter, a complete decontamination of 12 m^2 of the steam generator surface is to be considered as an assumption on the safeguarded side with respect to the radioactive substance release.

The assumed activity release from the primary circuit surfaces is conservative, similar to the leakage event at a main pipe. A retention in the building, for example, due to a plateout, has not been assumed. The assumed remobilization of radioactive substances sedimented on the steam generator surfaces by the interaction with water and steam has been proved for cessium by experiments conducted in Juelich NEP (B 144, B 145). In our opinion, the transfer of remobilization values for cessium to iodine and strontium is justified. The factor 0.5 used herein for J and Sr can be tolerated since the remobilization of the decontamination by water jet is dominated.

From our point of view, the 10 % fraction of the total surface area of the preheater as starting point for the interaction with water is conservative enough. The assumed period of water influence on the surface of one minute is also appropriate.

The governing specific activity in the graphite corrosion is experimentally proven with AVR-BE-pellets (B 147). The corrosion quantity of 300 Kg of graphite with 600 Kg of water was evaluated in Chapter 5.4.3. The corrosion quantity of 300 Kg of graphite deviates from the 240 Kg of graphite described in the note (D 3.3-2), since no closure of the blower cover is assumed. In this way results a mobilization by fuel element corrosion of 2.7 E+09 Bq of J 131, 2.1 E+08 Bq of Cs 137, 1.7 E+08 Bq of Sr 90, and 3.9 E+07 Bq of Ag 110 m.

Until 1986, the UO_2 oxidation had not been taken into consideration in the HTR safety analysis. Only after recommendations from the USA, where fission product release has been monitored mainly in cases of contact between steam and carbide fuels, this mechanism has been evaluated within the safety considerations.

Generally, the reaction displayed mechanism represents a possible release path. The knowledge of the evolving oxidation degree of UO_{2+x} dependent on the temperature, reaction time, and partial O_2-, CO- and H_2- pressures, is necessary for the purposes of an exact quantification. The question is to what extent a notable oxygen partial pressure occurred in the presence of graphite, that would be able to cause an UO_2 oxidation. Therefore, we consider that the applicant's estimate, which neglects the competitive reaction of graphite corrosion, is conservative. The oxidation degree assumed hereafter by the applicant, seems to correspond to the values identified in the available literature, mainly with respect to the simultaneous presence of H_2. The applicant's further release calculation is rooted in the measured (B 149 to B 151) diffusion coefficients of non-stochiometrical UO_2.

We based the hereby calculated release rates also in our continued estimate of the activity flow in order to deliver a fair example for the respective influence on the load dose to which the environment is exposed.

For the purpose of a more precise mechanism quantification, we consider important to examine closer the oxidation kinetics of the available multi-substance mix as well as the correlation between the oxidation and the release, especially with respect to the irradiated fuel.

It can be generally determined that, via this effect, a "massive" release is excluded, since only damaged particles can be affected, and that, even under the assumption of unfavorable release mechanisms, the planned guide numbers according to § 28 Section 3 Radiation Protection Regulation are not exceeded.

Due to the aforementioned evaluation of the calculation assumptions and individually carried out random checks we can agree with the applicant's nuclide vector applied to the release of radioactive substances from the primary circuit. With respect to the chemical composition and the iodine filter factor, we have made assumptions, which differ from those of the applicant, as described above.

Should iodine filtered through the secured low-pressure unit should be assumable, the iodine release rates would be reduces by two orders of magnitude.

Nuclide	Activity Release in Bq
Kr 83 m	2.8 E+8
Kr 85 m	2.7 E+9
Kr 85	3.4 E+7
Kr 87	5.0 E+8
Kr 88	3.6 E+9
Xe 131 m	1.1 E+8
Xe 133 m	1.1 E+9
Xe 133	3.4 E+10
Xe 135	9.8 E+9
J 131 el	6.5 E+ 8
J 131 org.	6.5 E+ 8
J 132 el	4.7 E+ 7
J 132 org.	4.7 E+ 7
J 133 el	2.9 E+ 8
J 133 org.	2.9 E+ 8
J 134 el	1.9 E+ 7
J 134 org.	1.9 E+ 7
J 135 el	1.9 E+ 8
J 135 org.	1.9 E+ 8
Sr 89	8.3 E+ 5
Sr 90	2.3 E+ 4
Ag 110 m	3.9 E+ 3
Cs 134	1.2 E+ 5
Cs 137	1.7 E+ 6

Table 5.8.1-4: Activity release through the chimney in the event of a leakage in the heating pipe of the steam generator in the preheater sector with pressure relief system activation

Nuclide	Activity Release in Bq
Kr 83 m	2.8 E+8
Kr 85 m	2.7 E+9
Kr 85	3.4 E+7
Kr 87	5.0 E+8
Kr 88	3.6 E+9
Xe 131 m	1.1 E+8
Xe 133 m	1.1 E+9
Xe 133	3.4 E+10
Xe 135	9.8 E+9
J 131 el	5.5 E+ 8
J 131 org.	5.5 E+ 8
J 132 el	2.8 E+ 7
J 132 org.	2.8 E+ 7
J 133 el	1.6 E+ 8
J 133 org.	1.6 E+ 8
J 134 el	1.1 E+ 7
J 134 org.	1.1 E+ 7
J 135 el	1.1 E+ 8
J 135 org.	1.1 E+ 8
Sr 89	9.0 E+ 5
Sr 90	2.5 E+ 4
Ag 110 m	3.9 E+ 3
Cs 134	2.5 E+ 5
Cs 137	3.5 E+ 6

Table 5.8.1-5: Activity release through the chimney in the event of a leakage in the heating pipe of the steam generator in the overheater sector with pressure relief system activation

5.8.1.5 Leakage in the Largest Primary Cooling Agent Line Outside the Reactor Building

The helium purification plant is located outside of the reactor building in an ancillary building. The most significant gas-driven radioactive substance release occurrs in the event of leakage of a unit connecting the primary circuit and the components of the helium purification plant. The safety report assumes a leakage or a break in a connecting unit between the primary circuit and the helium purification plant, located in the reactor ancillary building. A cutting off of the primary circuit is assumed with the primary circuit pressure of 50 bar. Gas discharge contributes to balancing the pressure of the helium purification plant with surroundings of the the leakage. According to the safety report, the released radioactively contaminated helium passes through the pressure relief valves of the helium unit into the reactor ancillary building to reach the outlet air chimney.

> The greatest potential of gas-driven gas-carrying radioactive substances into the reactor ancillary building is located in the helium purification plant, which is connected to the primary circuit. In the event of a regeneration of the helium purification plant, the same maximum inventory can be contained in a helium storage tank. Therefore, we agree with the applicant's opinion that, in the event of the postulated break between the primary circuit and the helium purification plant one has to deal with the radio logically most representative accident regarding gas-driven radioactive substances release outside the reactor building.

Activity Release

In the event of the break of a primary cooling agent in a reactor ancillary building primary circuit, radioactive substances are released, which stream out at pressure relief of 60 to 50 bar, together with the entire inventory of the molecular screen and the low-temperature absorber of the helium purification plant. The dust filter inventory of the helium purification plant remains unconsidered since no reverse flow can be assumed through this core filter due to the availability of a check valve. An occasional partial retention in the building has not been taken into account.

The applicant has summarized the radioactive substance inventories of both the molecular screen and the low-temperature adsorber of the helium purification plant and complemented them with the quantity of radioactive substances released, on the one hand, as primary gas fraction in the case of a primary circuit pressure reduction from 60 to 50 bar and, on the other hand, as non-absorption fraction from the surfaces, in order to determine the activity of the radioactive substances in the event of a break in helium purification plant. The dust filter inventory was not taken into account. Due to the availability of a check valve, we are in agreement with this. However, herefrom arise respective requirements to the tight seating of this valve. Based on the aforesaid, the postulated break is allocated to the pipe between the dust filter and the molecular screen. The conservative approach does not consider a retention of radioactive substances. In this manner, the radioactive substances mentioned in Table 5.8.1-6 reach the waste air chimney in unfiltered condition.

Nuclide	Activity Release in Bq
Kr 83 m	6.3 E+9
Kr 85 m	3.0 E+10
Kr 85	1.6 E+10
Kr 87	2.1 E+10
Kr 88	5.5 E+10
Kr 89	5.7 E+9
Kr 90	2.4 E+9
Xe 131 m	2.2 E+10
Xe 133 m	4.6 E+10
Xe 133	2.2 E+12
Xe 135m	3.8 E+9
Xe 135	1.2 E+11
Xe 137	9.8 E+9
Xe 138	2.0 E+10
Xe 139	3.0 E+9
J 131 el	6.8 E+ 6
J 132 el	9.0 E+ 7
J 133 el	4.4 E+ 7
J 134 el	2.3 E+ 8
J 135 el	7.7 E+ 7
Sr 89	1.6 E+ 4
Sr 90	4.3 E+ 2
Ag 110 m	3.8 E+ 3
Cs 134	5.1 E+ 4
Cs 137	1.1 E+ 5
Rb 88	2.6 E+ 9
Cs 138	9.9 E+ 8
H 3	5.6 E+12
C 14	6.0 E+10

Table 5.8.1-6: Activity release in the event of a break in the helium purification plant outside of the reactor building

5.8.1.6 Leakage in a Vessel with Radioactively Contaminated Water

According to the safety report, the applicant assumes the complete emptying of an evaporation concentrate vessel. The activity release into the power plant's surroundings is unfiltered and takes place with the waste air through the chimney.

The vaporization concentrate vessels are located in the reactor building and contain substance inventories of highest radioactivity. Therefore the assumption that, when there is a disruption there a complete drain of the evaporation concentrate vessel is correct.

Activity Release

In order to determine the air-driven activity consequent to the aforementioned disruption, the applicant has made the assumption that an vaporized fraction of 1 % would consist of a weight-related activity concentration of 5 % of the concentrate concentration. The air-driven iodine chemical form was assumed to be elementary.

The evaporation concentrate vessel inventory is shown both in the note (D 3.3-1) and in the safety report. The activity release assumed by the applicant is identical to the pressurized water reactor release as set forth in the calculation basics (B 14). With regard to the chemical appearance of the iodine, the applicant deviates from the above-mentioned calculation basics and tends to the conservative approach. The radioactive substances released with the waste air are shown in Table 5.1.8-7.

We have nothing against the activity release assumptions, in case the further operation of the evaporation unit should be excluded when there is a break in the pipeline inside the evaporator. This should be guaranteed within the detailed design prior to construction.

Nuclide	Activity Release in Bq
Co 60	4.4 E+ 7
Sr 90	9.3 E+ 4
J 131 el	1.6 E+ 4
Cs 134	2.2 E+ 6
Cs 137	2.8 E+ 7
Ag 110	1.9 E+ 4

Table 5.8.1-7: Activity release after a leakage of radioactively contaminated water from a container

5.8.1.7 Effects on the Reactor Ancillary Building

The systems in the reactor ancillary building are not designed earthquake-proof. Therefore, leaks at the activity-driving components and systems should be assumed as well as their direct release close to the soil.

The conservative approach assumes the failure of all components of the reactor ancillary building containing radioactive materials in the event of an earthquake. Another conservative assumption relates to the release of radioactive substances without any retention.

Activity Release

Based on the safety report and related thereto, the applicant draws on the close-to-the-soil release of radioactive substances in the event of an earthquake and covers activity inventories of the helium purification plant, the storage containments for radioactively contaminated helium, the wastewater intermediate storage vessels, as well of the evaporation concentrate vessels.

The inventories of the above-mentioned components are described both in the note (D 3.3-1) and the safety report. The applicant has covered all vessels containing relevant radioactive material in the reactor ancillary building. In other words, the components, which have not been explicitly mentioned, do not need to be taken into consideration.

Since the maximum inventory of the helium purification plant is contained in both the molecular screen and low-temperature adsorber or in the storage vessels for radioactively contaminated helium,

it is recommended to estimate the double value of the maximum inventory of the helium purification units and the storage containers for radioactively contaminated helium, since assumed inventory is contained a strand in the helium purification plant or the storage vessels.

Regarding the contribution of contaminated water, the same quantities of radioactive substances are released per evaporation unit as in the event of a leak of a containment for radioactively contaminated water. The availability of two evaporation circuits is taken into account by calculating the double assumed values.

In our opinion, the release of radioactively contaminated substances shown in Table 5.8.1-8 is representative for the event of an earthquake. The release of radioactively contaminated water from the reactor ancillary building into the soil has not been assumed by the applicant. We would join this assumption since it has been proven that the floor trough of a reactor ancillary building is designed to withstand earthquake effects.

Nuclide	Activity Release in Bq
Kr 83 m	1.3 E+10
Kr 85 m	6.0 E+10
Kr 85	3.2 E+10
Kr 87	4.2 E+10
Kr 88	1.1 E+11
Kr 89	1.1 E+10
Kr 90	4.7 E+9
Xe 131 m	4.4 E+10
Xe 133 m	9.2 E+10
Xe 133	4.3 E+12
Xe 135m	7.6 E+9
Xe 135	2.5 E+11
Xe 137	2.0 E+10
Xe 138	4.0 E+10
Xe 139	6.1 E+9
J 131 el	1.4 E+ 7
J 132 el	1.8 E+ 8
J 133 el	8.7 E+ 7
J 134 el	4.6 E+ 8
J 135 el	1.5 E+ 8
Co 60	8.9 E+ 7
Sr 89	2.2 E+ 8
Sr 90	1.9 E+ 5
Ag 110 m	3.8 E+ 4
Cs 134	4.6 E+ 6
Cs 137	5.6 E+ 7
Rb 88	5.1 E+ 9
Cs 138	2.0 E+ 9
H 3	1.9 E+13
C 14	1.2 E+11

Table 5.8.1-8: Activity release due to the effects of an earthquake on the reactor ancillary building

5.8.2 Calculation of the Radiation Exposure

Calculation Models and Parameters

§ 28, Section 3, of the Radiation Protection Regulation (B 2) establishes the maximum body doses that may be considered in the surroundings of a nuclear power plant during planning and design of structural and other technical protective measures against failures and accidents (accident planning figures). The accident calculation basics (B 14) of the directives of the Federal Department of the Interior on the design evaluation of nuclear power plants equipped with pressurized water reactors according to § 28 Section 3 RADIATION PROTECTION REGULATION contain, in their Chapter 4, calculation models and data sets for the irradiation exposure estimates. These calculation models consider the following exposure paths:

- the external radiation exposure caused by beta radiation consequent to an exposure in the waste air stream (beta submersion),
- the external radiation exposure caused by gamma radiation from the waste air stream (gamma submersion),
- the external radiation exposure caused by gamma radiation out of radio nuclides sedimented on the soil (gamma soil radiation),
- the internal radiation exposure caused by the inhaled radio nuclides (inhalation),
- the internal radiation exposure caused by radio nuclides, taken into the body with contaminated food (ingestion).

The models and data sets for the radiation exposure calculation of Chapter 4 of the Radiation Protection Regulation do not relate to any pressurized water reactor specific assumptions. They can be applied to accidental release cases of radioactive substances originating from HTR module power plants without restriction. Therefore, we have determined the radiation exposure caused by accidents based on these calculation models and radiological –environmental parameters.

The radiation exposure of adults caused by accidents has been calculated by us as a summarized dose over a period of 50 years. A young child is treated as a child in the first year following the accident and as an adult in the 49 following years. Differences between adults and young children derive merely from the irradiation dose of the first year following the accident. The accident calculation basics do not contain any specific regulation.

Due to the HTR module nuclear power plant concept, which is independent from any specific location, we have followed strictly the parameter sets of the accident calculation basics. After specifying the particular location it shall be determined if the data sets applied remain valid or if new location-specific parameters or additional paths should be taken into account.

Radiation Exposure in the Surroundings Caused by Radiologically Representative Accidents

In the above sections of Chapter 5, we determined the radiologically relevant accidents based on their effects on the environment. In Chapter 5.8.1 we summarized the release values of radioactive substances for those accidents. For every type of accident we furthermore determined if the release occurred close to the soil and if the retention effect of the filters should be taken into consideration.

Similar to the approach in Chapter 3.4.2 we determined a distance of 100 Km from the emission point as an unfavorable point in the nuclear power plant vicinity. Furthermore, according to the accident calculation basics we also estimated a critical distance of 2 Km.

In the background of these frame conditions and applied to the accident calculation parameters, we determined the short-term expansion factors, the fallout and washout factors, as well as the gamma submersion factors for the different exposure paths. The radiation exposure figures calculated by us regarding the radiologically relevant accident types are summarized in Tables 5.8.2-1 and 5.8.2-2.

It was shown that, with a single exception, the maximum dose values occurred in all the particular accident types at a distance point of 100 m. The effects of an earthquake on a reactor ancillary building showed the respectively highest radiation exposure body values as well as organ radiation exposure degree except for the thyroid gland dose. This applies both for adults and young children.

The highest thyroid gland radiation dose in adults and children is caused by an accident related to a core heating after a major pipe break (DN 65) in the case wherein the highest activity release rates occur within a period of 34 to 42 hours correlated with the most unfavorable expansion conditions (see Table 5.8.1-2, 2nd table). Since the dose values are generally very low, for the purpose of simplification, we based also the first time period of 0 to 34 hours on the most unfavorable expansion circumstances. The most unfavorable location in this case is located at a distance of 2 Km.

The maximum value determined by us for adults is 0.49 mSv and 0.28 mSv for children. The maximum organon dose is derived for the thyroid gland at 1.2 mSv for adults and 8.1 mSv for children.

For comparison, Tables 5.8.2-1 and 5.8.2-2 display the planned parameters for accidents according to § 28, Section 3, of the Radiation Protection Regulation. All the dose values are far below the respective accident planning figures. The radiation exposures calculated by us correspond generally with those presented by the applicant in the safety report as well as in the additional document (D 3.3-2). Therefore, we can generally confirm the dose calculations of the applicant.

The accident calculation basics do not specify any calculation procedure for H 3 and C 14 nuclides regarding the exposure path of ingestion. In the supplementary document (D 3.3-2), the applicant stated his reasons for not evaluating this path with respect to HTR module nuclear power plant.

In order to still evaluate the radiation exposure value ranges for H 3 and C 14 ingestion, we determined dose values according to SSK models (B 25). The H 3 and C 14 decay occurs solely in the event of a pressure relief phase caused by a major pipeline break (DN 65), pipe break in a helium purification plant, and the effects of an earthquake on a reactor ancilliary building (see Chapter 5.8.1). The H 3 and C 14 decay in the event of an earthquake are at their maximum point; furthermore, the nuclides are released close to the soil. For this reason, the radiation exposure by H 3 and C 14 ingestion in the event of the latter accident covers all other events.

For the effects of an earthquake, we calculated, based on SSK models (B 25), an additional body dose of approximately 100 μSv for adults and an additional bone portion (as highest organ dose) of approximately 190 μSv, as well as an additional thyroid gland dose

of approximately 100 μSv. For young children, we calculated for the whole body and all organs an additional portion of approximately 150 μSv. These values are to be added as highest estimate to the three events presented in Tables 5.8.2-1 and 5.8.2-2. Taking into consideration the H 3 and C 14 ingestion accident, the planning values set forth in § 28, Section 3 of the Readiation Protection Regulation are far below the stipulated limits.

This shall guarantee that in the course of planning the structural and other technical safety measures against accidents, the applicant has taken into consideration the required precautions based on the contemporary level of science and technology.

Similar to the approach presented in Chapter 3.4.2, we determined the portions within the radiologically representative accidents according to ICRP models 26/30 (B 22, B 23) with the help of the dose factors set forth by the Federal Health Department (BGA) (B 125, B 126). The results of our calculations are displayed in Tables 5.8.2-3 and 5.8.2-4. The differences between the dose values are evident merely in the applied dose factors; all the other calculation values are identical.

The radiation exposures displayed in Tables 5.8.2-3 and 5.8.2-4 differ slightly from the values shown in Tables 5.8.2-1 and 5.8.2-2. The dose values calculated according to ICRP 30/BGA are also far below the planned values as set forth in the most recent amendment to the Radiation Protection Regulation which will become effective in November of 1989.

Table 5.8.2-1: Radiation exposure in the environment by the emission of radioactive substances for radiological representative accidents, distance to the point of 100 m – Adults

	Accident Dose in µSv							
Accident	Bones	Liver	Whole Body	Thyroid	Kidney	Lung	Stomach / Intestine	Skin
Break in connecting line DN 65 Pressure relief phase	57	12	16	58	11	11	15	7.3
Core heating phase (1st version) without special consideration for the time interval 0-34 h	0.7	0.7	0.7	120	0.7	0.7	3.4	0.5
Core heating phase (2nd version) with targeted division of the time intervals 0-34 h, 34-42 h, etc.	4.1	4.0	4.0	730	4.0	4.4	24	2.8
Break in a measuring line	1.0	1.0	1.0	21	1.0	1.1	2.4	1.6
Steam generator heating pipe, near preheater	1.4	0.9	0.9	90	0.8	0.9	2.8	0.6
Steam generator heating pipe, near overheater	1.7	1.2	1.1	75	1.0	1.0	2.5	0.8
Break in pipeline of the He purification plant	2.1	71.	7.1	9.3	7.1	7.1	7.3	2.0
Drainage of the wastewater evaporator	57	50	46	45	45	42	46	41
Earthquake effects on the ancillary plant buildings	240	510	490	550	490	490	500	170
Core heating phase (2nd version) Distance 2 Km -	2.4	2.4	2.4	1200	2.4	2.6	25	0.4
Planning reference values of the § 28, Section 3 of the Radiation Protection Regulation	300000	150000	50000	150000	150000	150000	150000	300000

Table 5.8.2-2: Radiation exposure in the environment by the emission of radioactive substances for radiological representative accidents, distance to the point of 100 m – Small children

	Accident Dose in µSv							
Accident	Bones	Liver	Whole Body	Thyroid	Kidney	Lung	Stomach / Intestine	Skin
Break in connecting line DN 65 Pressure relief phase	56	9.5	13	280	8.0	7/6	8.1	7.3
Core heating phase (1st version) without special consideration for the time interval 0-34 h	2.2	2.6	1.6	760	1.0	1.6	0.6	0.5
Core heating phase (2nd version) with targeted division of the time intervals 0-34 h, 34-42 h, etc.	14	17	9.9	4700	6.1	9.9	3.8	2.8
Break in a measuring line	1.3	1.5	1.2	120	1.2	1.2	1.1	1.6
Steam generator heating pipe, near preheater	2.5	2.3	1.5	560	1.1	1.5	0.8	0.6
Steam generator heating pipe, near overheater	2.6	2.4	1.6	450	1.2	1.5	1.0	0.8
Break in pipeline of the He purification plant	1.5	3.6	3.5	16	3.5	3.5	3.5	2.0
Drainage of the wastewater evaporator	57	51	45	45	44	42	45	41
Earthquake effects on the ancillary plant buildings	230	300	280	600	280	280	280	170
Core heating phase (2nd version) Distance 2 Km -	19	22	13	8100	5.8	13	1.5	0.4
Planning reference values of the § 28, Section 3 of the Radiation Protection Regulation	300000	150000	50000	150000	150000	150000	150000	300000

Table 5.8.2-3: Radiation exposure in the environment by the radiological representative accidents, calculated with the dose factors according to ICRP/30 BGA, distance to the point of 100 m – Adults

	Accident Dose in µSv							
Accident	Effective Equivalent Dose	Red Bone Marrow	Bone Surface	Thyroid	Lung	Liver	Intestine	Skin
Break in connecting line DN 65 Pressure relief phase	9.0	11	15	37	7.6	7.4	7.9	7.1
Core heating phase (1st version) without special consideration for the time interval 0-34 h	3.3	0.5	0.5	94	0.5	0.5	0.5	0.5
Core heating phase (2nd version) with targeted division of the time intervals 0-34 h, 34-42 h, etc.	21	3.0	3.0	580	3.1	2.9	2.9	2.8
Break in a measuring line	1.4	1.0	1.0	14	1.0	1.0	1.0	1.6
Steam generator heating pipe, near preheater	2.9	0.7	0.8	73	0.7	0.7	0.7	0.6
Steam generator heating pipe, near overheater	2.8	1.0	1.0	61	1.0	0.9	1.0	0.8
Break in pipeline of the He purification plant	3.3	3.2	3.2	4.6	3.2	3.2	3.2	2.0
Drainage of the wastewater evaporator	46	46	47	46	46	47	47	41
Earthquake effects on the ancillary plant buildings	270	260	270	300	280	270	270	170
Core heating phase (2nd version) Distance 2 Km -	32	0.4	0.6	1000	0.7	0.5	0.5	0.4
Planning reference values for new edition of the Radiation Protection Regulation (11/89)	50000	50000	300000	300000	150000	150000	150000	300000

Table 5.8.2-4: Radiation exposure in the environment by the radiological representative accidents, calculated with the dose factors according to ICRP/30 BGA, distance to the point of 100 m – Small children

	Accident Dose in µSv							
Accident	Effective Equivalent Dose	Red Bone Marrow	Bone Surface	Thyroid	Lung	Liver	Intestine	Skin
Break in connecting line DN 65 Pressure relief phase	15	11	15	210	11	7.2	7.8	7.1
Core heating phase (1st version) without special consideration for the time interval 0-34 h	20	0.8	0.7	620	0.7	0.6	0.6	0.5
Core heating phase (2nd version) with targeted division of the time intervals 0-34 h, 34-42 h, etc.	120	4.8	4.4	3800	4.1	3.6	3.5	2.8
Break in a measuring line	3.7	1.1	1.1	88	1.1	1.0	1.0	1.6
Steam generator heating pipe, near preheater	15	0.9	0.9	450	0.9	0.8	0.8	0.6
Steam generator heating pipe, near overheater	12	1.1	1.2	370	1.1	1.0	1.0	0.8
Break in pipeline of the He purification plant	3.7	3.0	3.0	13	6.0	3.0	3.0	2.0
Drainage of the wastewater evaporator	46	46	47	45	46	46	46	41
Earthquake effects on the ancillary plant buildings	270	240	250	480	420	240	250	170
Core heating phase (2nd	210	3.7	3.0	6700	2.4	1.7	1.5	0.4

version) Distance 2 Km -								
Planning reference values for new edition of the Radiation Protection Regulation (11/89)	50000	50000	300000	300000	150000	150000	150000	300000

6 Decommisioning

In his safety report the applicant describes three decommissioning strategies, as follows:

A Safeguarded Enclosure

Total dismantling of the fuel elements, primary cooling agent, and radioactive waste such as filter elements, wastewater, concentrate, etc. All the remaining solid radioactive materials are enclosed in a safe enclosure.

B Partial Dismantling Combined with a Safeguarded Enclosure of the Remaining Elements

In contrast with the safe enclosure, part of the radioactive and non-radioactive construction elements and components are safely recycled or removed.

C Total Dismantling

All radioactive and non-radioactive plant elements are dismantled and removed from the plant. The radioactive elements are transferred into a final radioactive waste facility.

A combination of the above strategies is possible, according to the applicant's presentation. However, the study neither covers nor provides any detailed information on the planned time schedule for concluding the work nor on the type and succession of the dismantling activities to be carried out under scenario C. In the aftermath of the designed reactor lifetime of 32 years, the core radioactivity level is outlined in the safety report at 50 mSv/h, where Co-60 and Ta-182 determine the decay level.

The following evaluation criteria have been implemented in order to assess the decommissioning activities presented by the applicant:

- Radiation Protection Regulation /B 2/

§ 28 Section 1	Radiation Protection Principles (Minimization Decree)
§ 45	Radiation level limits outside of the radiation safety areas
§ 49	Radiation dose limits for radiation personnel expose to radiation
§ 52	Incorporation of radioactive elements

- Safety Criteria on Nuclear Power Plants /B 6/
  Criterion 2.10: Decommissioning and Dismantling of Nuclear Power Plants

- RSK Guideline on Pressurized Water Reactors /B 10/
  Chapter 16 Decommissioning and Dismantling

Although the latter guideline has been developed for pressurized water reactors, the requirements of Chapter 16 also apply logically to high-temperature reactors.

The removal of fuel elements, waste, etc. prior to initiating the actual decommissioning activities is conducted as during the plant operational phase and therefore does not cause any specific problems.

Relevant experiences have been gained already in the Federal Republic of Germany on the inception of a safe enclosure (Strategy A or B). For instance, a safe enclosure of the Lingen nuclear power plant was already

carried out. The respective work in this decommissioning phase is primarily concentrated on peripheral, slightly contaminated plant parts. Based on their technical implementation and with regard to the radiation protection issues, they are comparable to the repair, reconstruction, and decontamination activities during plant operation. Therefore, the experience gained proves the possibility complying with all the radiation protection guidelines during this phase.

The complete dismantling of an entire nuclear power plant designed for high-power performance and an extensive lifetime, however, has not yet been attempted in the Federal Republic of Germany. Still, some studies /B 152, B 153/ on international experiences guarantee the possibility even for a safe removal of highly activated plant components of nuclear reactors operating in Germany today with respect to the valid radiation protection regulations. This statement is also applicable to the HTR 2-module reactor, based on the outline below.

Smaller components are also dismounted and transported within the frame of repair work during the plant operation phase, whereby those are fit into easily accessible areas for reasons of operational radiation protection and in the normal case are transported as single parts /D 1/. The requirement of the RSK Guideline /B 10/, Section 2, of laying and arranging the components is consequently also met in this case.

Larger components, such as the reactor pressure vessel and the steam generator shall be taken apart into pieces in a final repository. These components, together with the dry cooling tower and the reactor cell wall, are at the same time the components containing the major activity share.

The construction of the components, the neutrons flow during operation and commissioning are approximately similar to the conditions in the

reactor pressure vessel (RDB) or THTR 300 (reactor pressure vessel mounting). Therefore, all the work required for the introduction of a safe enclosure or components removal and transport are characterized by the similar quality. This also applies to the gas circulator and reactor cell wall.

With respect to the maximum decrease of the radiation contamination of the personnel during the course of dismantling work of the plant and planning that is different from the current planning /D 1/, we recommend to check the possibility and potential benefit of a two-part reactor cell wall based on the instructions of the RSK Guideline, Section 1, as applied in some pressurized water reactor plants.

In order to reduce the radiation exposure to the minimum, it should be generally beneficial to postpone the exposure-intensive work until dominant nuclides have decayed in their activity level. The significant contribution of Co-60 and Ta-182 to the dose of 50 mSv/h in the reactor core indicated by the applicant appears to be realistically assessed, so that a safe enclosure prior to dismantling is considered to be reasonable.

The following measurements are basically suitable to significantly reduce the radiation exposure according to § 28 Section 1. Therefore, we recommend an additional check of the technical feasibility and effectiveness of these actions based on the actual conditions within a HTR 2-module nuclear power plant.

- The use of cobalt- and tantalum-poor materials in areas exposed to high neutron flow in order to reduce the origination of the above mentioned dose intensive radionuclides.

- The selection and optimization of construction elements and materials appropriate to the exposure to powerful neutron flow in order to avoid activation product of high durability. For instance, the emerging of Ca-41 (half-decay period of 100 000 years) can be reduced by the use of calcium-poor concrete additives in core construction. It is recommended to also reduce also the fraction of trace elements. In the course of the neutron activation, these can develop to long-lived or dose-intensive radionuclides. The use of boron-added concrete in the construction of the reactor cell wall can be reduced the activation product fraction in concrete /B 154/.

In general, it can be concluded that the plant's decommissioning and dismantling is possible with respect to the above-mentioned instructions. Further details shall be determined during the course of the design and prior to the decommissioning of the plant.

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Compliance with the Conventional Regulations
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Dimensioning Calculation for the Pressure-carrying Enclosure
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Stress and Load Spectrum for a Leak-before-Break Proof on the Connecting Pressure Vessel
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Loading of the Pressure Vessel Unit with Malfunction of the Impulse Brake
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Testing of the Mechanical Break as to the Leakage-before-Brake Behavior of the Feed Water Nozzle on the Steam Generator Pressure Vessel
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Pressure Vessel Unit
Testing of the Mechanical Break as to the Leakage-before-Brake Behavior of the Fresh Steam Nozzle on the Steam Generator Pressure Vessel
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Average Loads Fuel Element Nozzles

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State of the Qualification of the Materials Intended for Use in the High Temperature Range
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Pressure Vessel Unit
Concept for Recurring Tests:
Compliance with the Requirements of KTA Regulation 3201.4 and the Conventional Regulations
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Concept of Recurring Non-Destructive Tests on the Components of the Pressure Vessel Unit of the HTR Module
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/U 2.6.2-30/
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Recurring and Initial Pressure Tests on the Pressure Vessel Unit
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/U 2.6.2-31/
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Feasibility of Recurring Non-Destructive Tests on the Components of the Pressure Vessel Unit of the HTR Module
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Siemens / Interatom
Testing of the Mechanical Break as to the Leakage-before-Brake Behavior of the Connecting Pressure Vessel
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External Loads and Pipe Break Loads on the Feed Water Nozzle and the Fresh Steam Nozzle of the Steam Generator Pressure Vessel
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State of the Qualification of the Materials Intended for Use in the High Temperature Range
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/U 2.6.2-35/
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Stress Determination on the Fresh Steam Nozzle for the Testing of Mechanical Break
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Testing of the Mechanical Break for Excluding a Break of the Fresh Steam Nozzle on the Steam Generator Pressure Vessel
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/U 2.6.3-1/
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Operational Experiences with Component Test Circuit for High-temperature Reactors
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Steam Generator of the HTR Module Power Plant
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Long-term Stability Parameters for Incoloy 800
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Failure Assumptions in a Break of the Pipe of the Steam Generator
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Primary Circuit Blower
Possible Oil Penetration into the Primary Cooling Agent Circuit
ID No.: 76.00633.4 of 05/27/88

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Primary Circuit Blower for the HTR Module Power Plant – Supplementary Description
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Primary Circuit Blower
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Limitation of the Fresh Steam Temperature
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/U 2.6.8-1/
Siemens / Interatom
Leakage Detection on in the Pressure Vessel Unit
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/U 2.6.8-2/
Siemens / Interatom
Leakage Detection on the Fresh Steam and Feed Water Nozzles of the Steam Generator Pressure Vessel
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/U 2.6.9-1/
Siemens / Interatom
Pressure Relief System Primary Circuit
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Primary Circuit Check Valves
Classification, Structural, and Functional Features
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/U 2.7-1/
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Description of Helium Purification System and the Helium Secondary Systems
ID No.: 76.00530.1 of 11/16/87

/U 2.7-2/
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Siemens / Interatom
Description of the Operational Experiences with the Main Components of the Fuel Element Filling Device of AVR
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Evidence of the Compliance for Fuel Element Design Limitations in the Equipment, Taking into Consideration a Reloading and an Outage of the Brake Gas
ID No.: 76.00603.3 of 07/07/88

/U 2.7-5/
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ID No.: 76.00680.7 of 10/12/88

/U 2.9-1/
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Concept of their Task and Function
ID No.: 76.00722.6 of 02/28/89

/U 2.9-2/
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Concept of their Task and Function
ID No.: 76.00721.4 of 02/28/89

/U 2.9-3/
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Fresh Steam Safety Valve
Concept of their Task and Function

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Steam Generator Relief Valves
Task and Function
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Arrangement according to the Principle of the Feed Water and Fresh Steam Valves in the Reactor Building
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Effects of the Concept Changes Regarding the Fresh Steam Valves and the Secondary Pressure Safeguard in Disruption Analyses
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Long-term Failure of the Surface Cooler
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Description and Function of the Emergency Control Room of the HTR Module Power Plant
ID No.: 76.00541.7.A of 12/08/88

/U 2.12-3/
Siemens / Interatom
Preliminary Power Determination of the Emergency Electric Power Plant
ID No.: 76.00681.9 of 09/29/88

/U 2.12-4/
Siemens / Interatom
Safety Concept in Case of a Failure of the AC and DC Current Supply
ID No.: 76.00624.2 of 08/18/88

/U 2.12-5/
Siemens / Interatom
Fire Protection Concept
ID No.: 76.00666.0 of 09/29/88

/U 2.12-6/
KWU
Rough Arrangement Plans with Essential Data for Components, Cable, and Pipeline Tracks
(Reactor Building, Switching Station, and Emergency Supply Building)
Attachment to the Letter to TUV Hanover e.V. of 10/09/87

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Siemens / Interatom
Pressure Glass Lead-Throughs on the Reactor Pressure Vessel and on the Steam Generator Pressure Vessel
ID No.: 76.00693.6 from 05/05/88

/U 2.13-1/
Siemens / Interatom
Concept Description of the Reactor Protection System
ID No.: 76.00634.6 of 08/12/88

/U 2.13-2/
Siemens / Interatom
Decoupling Concept of the Control System
ID No.: 76.00683.2 of 09/27/88

/U 2.13-3/
Siemens / Interatom
Primary Cooling Agent Pressure as Second Reactor Safety Stimulation Criterion in the Disruption with a Steam Generator Leakage
ID No.: 76.00690.0 from 21.11.88

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Behavior of the Reactor Protection System of the HTR Module in the Event of an Aircraft Crash and an Explosion Pressure Wave
ID No.: 76.00540.5.A of 10/14/88

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ID No.: 76.00599.* of 06/21/88

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Influence of the Particle Break Fraction by:
- Erroneous Bandwidths of the Input Data to Determine the Maximum Fuel Temperature
- Statistical Uncertainties in the Combustion Measurement

ID No.: 76.00692.4 of 11/22/88

/U 2.13-8/
Siemens / Interatom
Reliability of the Electric Supply for the Important Safety-Relevant Systems
ID No.: 76.00677.6 of 09/21/88

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Referring to Possibilities for the Differentiation of Different Pebble Types during First Loading
ID No.: 76.00708.* of 02/22/89

/U 2.15-1/
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Structural and Plant Engineering-related Fire Protection Measures, Escape Roads
ID No.: 76.00580.*.A of 02/22/88

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Siemens / Interatom
Determination of Fire Loads
ID No.: 76.00698.5 08/04/88

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Room Inventories for Reactor Buildings (UJA), Reactor Building Extensions (UJH), Reactor Ancillary Buildings (UKA), and Switching Station and Emergency Buildings (UBR)
ID No.: 76.00697.3 of 09/30/88

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Siemens / Interatom
Shielding and Local Contamination Rates
Explanations to Chapters 3.4.6.2 and 4.1.4 of the Safety Report
ID No.: 76.00589.6.A 05/20/88

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Siemens / Interatom
Activity Losses during Maintenance Work
ID No.: 76.00625.4 of 07/18/88

/U 3.3-1/
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Activity Emissions and Radiation Exposure in the HTR Module Power Plant
Part 1: Normal Operation
ID No.: 76.00560.2.A of 11/15/88

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Activity Emissions and Radiation Exposure in the HTR Module Power Plant
Part 2: Disruptions
ID No.: 76.00561.4.A of 11/15/88

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Reasons for the Conditions of the Secondary Water-Steam Losses of 4:1
ID No.: 76.00678.8 of 10/16/88

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Activation Products in the Water-Steam Circuit
ID No.: 76.00661.1 of 10/03/88

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ID No.: 76.00659.2 of 07/04/88

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Decision Criteria and Description of the Release Process out of the Storage Containers for Radioactively Contaminated Helium
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Siemens / Interatom
Letter to TUV Hannover e.V. 02/22/89
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Monitoring of Radioactivity in the Ambient Air
ID No.: 76.00626.6 of 07/21/88

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KWU
Regulation Schematic
Appendix to a Letter to TUV Hannover e.V. of 05/19/88
U5 251/908 508/0816/Be/Bo

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Curves of the Temperatures and Throughputs with a RESA, Failure of the Feed Water, Emergency Power Supply, Helium Blower
Plots delivered during an expert conference in Bensberg on 03/16/88 and 03/17/88

/U 4.2-2/
Siemens / Interatom
Natural Circulation in the Primary Circuit
Plots delivered during an expert conference in Bensberg on 03/16/88 and 03/17/88

/U 4.2-3/
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After-heat Discharge of the HTR Module (incl. Annex)
ID NO.: 76.00536.2 of 11/13/87

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Recurring Tests of the Electric Own Supply Plant
ID No.: 76.00682.0 07/19/88

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Temperature Development in the Core, the Graphite Built-in Components, and the Pressure Vessel after a Failure of the Main Heat Drain, with Intact Surface Cooling System, with and without Pressure Relief
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Temperature Development in the Core after Failure of the Main Cooling System
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Method for the Heat and Flow-related Analysis of the HTR Module with a Heat Relief Exclusively via Surface Cooler
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Bypasses in the Ceramic Core Built-in Components of the HTR Module Power Plant
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Temperature Distribution for the Initial Condition at105 % Performance, 280° C Cold Temperature, and 750° C Heating Gas Temperature
Appendix to a letter to TUV Hannover e.V. of 12/07/88

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