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The Karlsruhe Reference Design of a 1000 MWe Sodium Cooled Fast Breeder Reactor

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A. Introduction

As it has been described earlier [1] the present phase of the Karlsruhe Fast Breeder Project extends to 1967 and is aimed towards the investigation of the feasibility of 1000 MWe fast breeder reactors. One principal coolant being considered is sodium. The purpose of the present Karlsruhe study [2], called Na-1, of a 1000 MWe sodium cooled breeder reactor was to serve as a reference design. On the basis of this reference design the frequently discussed competition between breeding, safety and economy is being explored in a fashion which is as much quantitatively as possible. It is in particular the impact of safety on breeding and economy which is being studied. A first set of results has been presented yesterday by Smidt [3].

The proper choice of a design basis accident (DBA) and its impact on capital costs is being explored right now. All this will serve as the basis of a Na-2-study. It is therefore this forthcoming Na-2-study instead of the present Na-1-study which could serve as a basis for a thorough safety evaluation in the style of a safety report.

It is the purpose of this paper to describe the present Na-1-design and its safety features with respect to incidents, the maximum hypothetical accident (MHA) and the DBA, in view of the forthcoming Na-2-study.

B. A Description of the Karlsruhe Na-1-Reference Design

A detailed description of the Na-1-design is given in a Karlsruhe publication [2]. It also has been described for the recent Detroit Conference [4]. The descriptions should be considered as a background for this chapter.

The table of the Apprendix gives the most important data of the Na-1-design.

1. Choice of Fuel

The fuel for Na-1 are the mixed oxides UO_2/PuO_2. The design objective was to arrive at fuel cycle costs below 1 mill/kWh. This can be met without difficulties by the use of the oxides [5]. The advantages of using other types of fuel such as the carbides whould lie therefore in the areas of breeding including the aspect of lowering the fuel cycle costs or capital costs. We will touch that point later. The choice for UO_2/PuO_2 as a fuel is consistent with the Karlsruhe approach of also looking for H_2O-steam cooling. This coolant excludes the use of the carbides. Commonly realized is the advantage of having available the technology of UO_2 as a fuel. The technology of UC is less developed and somewhat more difficult. Realizing the fact that this fast reactor fuel development is the first large scale fuel development at Karlsruhe the advantage of the better known and more easy technology of UO_2/PuO_2 was compelling.

One further argument for the oxides should be mentioned because it is used only seldomly: Suppose that λ is the heat conductivity of the fuel in the rod of diameter R and α is the heat conductivity in the gap. Then the "effective" heat conductivity λ_{eff} acting along the temperature difference: center of fuel - cladding is

$$\lambda_{eff} = \lambda \frac{2\alpha R}{\alpha R + 2\lambda}$$

If $\frac{\alpha R}{2\lambda}$ is small, then $\lambda_{eff} \rightarrow \alpha R.\frac{\alpha R}{2\lambda}$ for a fast reactor fuel rod with carbide fuel is small for three reasons: 1) R of a fast reactor rod is small, the situation is different for thermal reactors, 2) the problem of inner corrosion between fuel and clad is more serious for the carbides than for the oxides and afflicts the value of α, 3) the high value of λ in case of the carbides emphasizes the difficulties of getting a comparable conductivity $\frac{\alpha R}{2}$ in the gap.

2. Choice of Cladding and Structural Materials

In the four recent 1000 MWe studies austenitic stainless steel with preference to SS 316 and 304 respectively has been chosen as cladding materials. In the Na-1-study we took Incoloy 800 as a reference point for advanced Incoloy-types. On the other hand SS 16 13 CrNi as basic alloy for high creep strength stainless steel is alternative and might possibly be used in the Na-2-design. These two types of alloys seem to us to be the most promising, more than the above mentioned 316 and 304 types. We expect these relatively low Ni-alloys to have on one side an acceptable corrosion-behaviour respecting mass transfer and on the other side not to be harmful with respect to the neutronics. The neutronic evaluations with our first choice Incoloy 800 having the larger Ni-content of the considered choices showed that there is no great neutronic penalty for the use of such alloys.

The sodium temperatures are lower than in the American studies. The mean Na outlet temperature is 580°C (instead of 600-650°C in the US studies). If the hot channel factor is included the maximum outlet temperature of sodium is 630°C in our study. The mean inlet sodium temperature of the steam-generator amounts then to 560°C thus leading to a steam temperature of 540°C. The sodium outlet temperature of the steam-generator was taken as 360°C, the inlet temperature into the core will be 430°C.

Having fixed these temperatures one now can make a proper choice of the structural materials. The first choice is,of course, austenitic steel on the hot side of the loops: X 8 CrNiNb 1613 is the only stainless steel which is allowed for high temperature use in Germany. But for economic reasons we would prefer SS 3o4 or 316 and will discuss this question with our supervising authorities to get possibly the license for this material. We are also taking into consideration the Nb-stabilized ferritic steel 1o CrMoNbNi 91o advocated by the INTERATOM-group. We expect further economic gains using this steel in case it will prove to be successful. Furthermore we think it would be the most economic solution to use normal unstabilized ferritic steel 1o CrMo 91o for all cold parts of the loops with temperatures below 45o ^{o}C. As you will see even the reactor vessel might be manufactured using this relatively cheap steel and also for the concentric in- and outlet tubes to the heatexchangers this cheap ferritic steel may be used. The most difficult part of the sodium circuits with respect to materials are surely the tubes of the steam-generator for which we have no simple solution at hand. We think of a combined solution of normal ferritic steel for the evaporator section and austenitic steel for the superheater part of the steam-generator. (In this point the data of our study are misleading. Not an alternative but a combination of the austenitic and ferritic steel is intended.)

3. The Core Design

Originally a number of rather inconventional core designs were considered, for example spherical fuel elements of canned oxide fuels. But after considerable discussion we arrived at a design with conventional pins in an hexagonal subassembly. The basic argument was over all simplicity. The core has two zones of two enrichments optimized for power flattening. This results in two core zones of the same volume. It gives automatically also a good Doppler coefficient ($7 \cdot 1o^{-6}$ at 14oo ^{o}C) because the center zone has the lower enrichment. The

Na-void effect is acceptable. This point will be touched later. After considerable parametric studies the H/D ratio was chosen to be 1/3. It did not appear to be necessary to go to more extreme values or other extreme solutions in order to enhance leakage. A H/D value of 1/3 does not decrease the Doppler coefficient too strongly and also does not influence the fabrication costs of the fuel cycle too much. As will be pointed out in chapter 7. a core of a low H/D value may lead to somewhat lower values of the kinetic energy release in case of a major accident. The fission gases are given to a gas plenum at the lower end of the core. Therefore we have a vented to plenum scheme. The aspects of a vented to coolant scheme are evaluated in a paper of Sommer and Smidt here at this conference [6]. There seem to be no major economical advantages but an improvement of safety. We come to this point later. The radial blanket has also two regions, the outer region uses U-metal in order to reduce the size of the radial blanket (4o cm), but for reasons of the fuel cycle this appears not to be a good solution. This leads to the core arrangement shown in fig. 1.

The reason for having the plenum for the fission gases at the lower end is the lower sodium temperature. This leads to 8o cm length of plenum and saves about 25 cm of plenum length. Therefore the height of the subassembly is reduced and also the pressure drop of the sodium along the subassembly is smaller and requires therefore somewhat less structural material. Besides of that the thermal stresses in the subassembly are lower because the plenum does not participate in the displacement of the upper parts of the subassembly. The coolant fraction (α) in both core zones is 5o $^{o}/o$. The leading argument for that choice was the rather low value of the pressure drop in the core ($4.25 \frac{kp}{cm^2}$) which in turn reduces the necessary volume fraction for the structural materials (1o $^{o}/o$). The variation of the coolant fraction has been discussed at a different place [3].

The fuel subassemblies are fixed in a lower grid plate, the hold down is hydraulic, the direction of coolant flow is upward. Originally the direction of coolant flow was discussed at length, but among other reasons the direction of natural convection and the pressure of the cover gas lead to the upward flow.

Much attention was paid to the question of refueling (see fig. 2a - c). There are three loading devices in region above the core, each covers 1/3 of the core and blanket area. Normal operation takes place under sodium and makes use of power steering. As long as everything is in order, quick loading and unloading is easily achieved therefore. In case of a mechanical failure the principle of power steering immediately detects this. Now it is possible to lower the level of the sodium pool and to have optical access to the upper core surface by a periscope device. Appropriate measures are now possible. Lowering of the Na level is possible because the Na pool above the core is a mixing chamber. Presently 9o $^{o}/o$ of the outcoming flow of the core is taken away from the outlet nozzles by thin vertical plates, only 1o % of the flow reaches the outlet point directly. This decreases the effect of temperature shocks to the sodium containing walls. And the vertical plates act as walls in case the Na level is lowered, therefore sort of a under flow hot cell with optical access by a periscope device is established in this case. The unloaded subassemblies are given to waiting and transfer baskets at the outer perimeter of the radial blanket, from there they are taken away after a cooling-off period of 1oo days by a rather simple device.

The number of three loading devices might lead to very short interruptions for refueling. As long as more than one loading per year appears to be necessary this may be of great importance for the question of availability. In our case we have a rating of 1 $\frac{MW_{th}}{\text{kg fiss mat}}$ and a three batch loading scheme leading to refueling periods of 196 days. This is reasonable but not yet in coincidence with the yearly maintenance cycle and therefore very short interruptions for refueling are desirable.

It is probably more important to look for refueling periods of one year rather than for high ratings if they are associated with short refueling periods. The fuel cost benefit of such high ratings may be obtained only if a power refueling appears possible. This possibility should not be excluded for future reactor studies, however.

A refueling period of one year could increase the economic potential of the plant. Such a period can be obtained with high values of the excess reactivity, this at least in most cases hurts breeding and economy except the internal breeding ratio is somewhat larger than one.

In our case the excess reactivity for bridging a burn-up of $\frac{100.000}{3}$ MWd/to is 2.4 $^{o}/o$. This value is the sum of Pu burn-up (0.75 $^{o}/o$) and fission product poisening (1.65 $^{o}/o$), the internal breeding ratio is 0.89 (inner zone 0.556, outer zone 0.334).

The nuclear core calculation use an equilibrium composition of Pu isotopes [7]. This equilibrium composition is such, that only that amount of freshly bred Pu from the blankets goes to the core, which is necessary for making up the Pu burn-up in the core. This with all probability will not be the scheme for Na-2. It is intended to have a mixed core-blanket aqueous reprocessing scheme instead. After one refueling period, $\frac{1}{n}$ of the core and $\frac{1}{2n}$ of the radial blanket would be mixed together in the process line. If the internal breeding ratio is in the neighbourhood of one, this leads to one unique Pu isotope composition at all stages of the fuel cycle and there would be no Pu hold up in the radial blanket.

4. The Design of the Primary Circuit

The primary circuit is of the loop-type. Having a large reactor, a pool type might have serious problems, such as unforseen flow patterns, thermal stresses in the structural plates for the pumps and the heat exchanger, radial expansion of the components in the pool with respect to the concrete shield plug, difficulties of having easy access for maintenance and for heavy instrumentation and others. A solution of the loop-type can be better overlooked and the design is more straight forward. But it is not excluded that ultimately the pool type is the right choice.

The loop type solution of Na-1 provides two parallel circuits, that means two intermediate heat exchangers. Each circuit has two pumps in parallel which makes a total of four. A higher number of heat exchangers does not seem advisable because this component should be a reliable one and a higher number of heat exchangers could perhaps even increase the failure probability. In the same sense the number of four for the pumps is assumed to be something like an optimum. The suspensions of heat exchanger together with the pumps provide for radial expansion thus going into the direction of a good type feature. All components of the primary circuit are in chambers of a concrete structure. The heavy motor drives for the Na-pumps are above the main floor, that means they are borne directly by the concrete structure, whereas the pumps themselves are directly connected with the intermediate heat exchanger. The reactor tank itself has a double wall in order to guarantee always the sodium level is above the reactor core. Convection from the reactor vessel to the intermediate heat exchanger provides a coolant flow large enough to pick up about 7 % of the nominal heat production. As a first order approximation it was assumed that there will be no main primary pipe failure. Together with the cooling capability of natural convection this eliminates the necessity of an emergency cooling system. As we will see later on too, the Na-1-design does not make too pessimistic assumptions in order to study the capital cost potential of this kind of a reactor. The systems and safety analysis of Na-1 may well lead to the necessity of having an emergency cooling circuit for the coolant of the reactor vessel of the Na-2-study.

All main pipes are co-axial. Therefore it is possible to have all outer pipe walls and all the containers at the same temperature, namely that of the sodium entrance, 43o °C. On the other hand this leads to great difficulties in having gate valves at the exit of the reactor vessel. Consistent with the assumption of not having a main pipe break no gate valves were provided in the Na-1-design. This is also subject to a careful re-evaluation for Na-2.

The reactor tank is located excentrically in the containment building in order to have a small diameter (28 m) of the reactor building.

The secondary circuit leaves the reactor building, steam-generators and turbines are outside.

The cover gas of the primary circuit is Argon.

Generally one can say that advantage is taken by the fact of the low pressure drop of the primary circuit (5 kp/cm^2). This together with the fact of the rather low constant overall temperature (43o ^{o}C) for vessels, containers and the outer wall of the main pipes leads to the possibility to use perhaps a cheap ferritic material for these components. The 2o MWe KNK-reactor at Karlsruhe is designed along this line of arguments.

5. The Neutronic Evaluation

The neutronic evaluation was done in several steps. First a parametric survey study was performed in calculating a large number of one-dimensional - one core zone reactor configuration. The results of these one-dimensional reactor calculations were checked against 8 two-dimensional - one core region reactors which fit thermodynamically in the scope of the design. The calculation of this stage used a slightly modified 16 group YOM set for the one-dimensional calculations; for the 8 two-dimensional calculations a condensation into 5 groups was necessary for reasons of computer time. In the process of condensation different weighting spectra were used for the core, the axial and the radial blanket. These spectra were obtained by one- and two-dimensional calculations and the results obtained after the condensation were cross checked and appeared to be satisfactory.

A second step was the neutronic calculation and optimization of the two region core. Again the 16 group YOM set, always in a slightly modified version, was used. The optimization was done with respect to the power flattening using proper enrichment in the two zones. This optimum appeared, when the two core regions were of the same volume. These core dimensions and enrichments were used for the reference reactor. As mentioned before, the Pu isotopic composition was that of an equilibrium core with separate core-blanket management.

Having these first datas for the reference reactor at hand a great effort was started to calculate this reactor very carefully. The results which were at first calculated with a one-dimensional code were cross checked against a 5 group two-dimensional S-2-calculation. The agreement was quite good. Thereafter the calculation was repeated with the 26 group Russian set because this group set had all materials of interest available as selfshielded, temperature dependent group constants, including fission products. During that step also the details of the flux distribution in particular in the radial blanket were considered.

The third step was to calculate the power and energy coefficients. The Doppler coefficient was calculated by means of the Karlsruhe code DCP (Doppler Coefficient by Perturbation) using as spectrum the results calculated with the 26 Russian group constants set (ABN). The neutron life--time and β_{eff} was calculated on the basis of the 16 group modified YOM set. The Na-void effect was calculated again using a two-dimensional code "Twenty Ground" and 6 neutron group coming from a condensation of the Russian 26 group set. The results were compared, with the corresponding one-dimensional results. The difference of the two maximal Na-void effects was less than 1o $^{o}/o$ (compare table 1 + 2).

Table 1: Na-void effects

(One-dimensional calculation by using the ABN-set)

	Void (1)	Void (1+2)	Void overall	Void max.
Δk	0.01o32	0.oo641	0.oo5445	0.o11747
ς	2.87 \$	1.78 \$	1.655 \$	3.27 \$

(1) and (2) referring to zone 1 and 2, (max.) to $r < 118$ cm.

Table 2: <u>Na-void effects</u>

(Two-dimensional calculations by using a 6 group set coming from a condensation of the ABN set)

	Void (1)	Void (1+2)	Void overall	Void max.
Δk	0.00654	0.00375	- 0.00479	0.01108
ρ	$1.8104\ \$$	$1.037\ \$$	$-\ 1.326\ \$$	$3.066\ \$$

One should note the difference in sign for the overall effect. This throws a light on the calculational difficulties which still exist.

Much attention was paid to proper burn-up calculation. The basis was a model for batch loading. Local groups of n subassemblies were formed. Each subassembly of these local groups belongs to a different batch. Unloading of a subassembly appears at the neighbourhood of 100.000 MWd/to. If the number of batches is n, the average burn-up in the local groups varies only for $\frac{100.000}{n}$ between two successive refueling steps. A choice was made to have n = 3 in the inner zone and the inner region of the outer zone and n = 4 in the outer region of the outer zone. This leads to a fuel lifetime of 3 · 196 d and a reactivity change of 0.7 °/o due to Pu burn-up, if fission products are included it will be 2.5 °/o.

Table 3: <u>The effects of fuel burn-up for various numbers of batches</u>

Number of batches in the inner and outer zone of the core	Δk_{eff} without fission products	Δk_{eff} fission products included	Average to maximum burn-up **)	Time difference between successive refueling steps
inner zone, outer zone n=3	0.76 °/o	-	79 °/o	196 d
inner zone: n=3; outer zone: inner region n=3, outer region n=4	0.74 °/o	2.5 °/o	85 °/o	

*) Calculation by using the ABN set

**) averaged over the radius r

The choice of n = 3 and n = 4 in the outer zone of the core instead of n = 3 in the whole core increases the average to maximum burn-up by 6 %.

Finally reactivity values were calculated on the basis of a perturbation theory for the interchange of a subassembly of the outer with inner zone and for the Na-voiding of the central subassembly.

The described procedure lead to the values published in [2], but the calculation of the reference design is being improved continuously. The new results are calculated with the KFK-26-group constants set described by H. Küsters in this conference [9]. This set has the same group structure as the ABN set but starts from different microscopic data and uses a typical spectrum of a sodium cooled 1ooo MWe reactor as weighting spectrum. The main changes in the results are the following:

critical mass is increased by about 6 %
the Doppler coefficient is decreased by about 25 %
the maximum Na-void reactivity value increases from about 3 \$
to about 7 \$.

6. The Accident Analysis

The accident analysis for the Na-1-design is now under way. It is not included in [2], only a very first version is indicated there. First results of the accident analysis are reported by Smidt et alii during this conference [3]. Therefore this chapter refers explicitly to this presentation.

The basic approach for the accident analysis comes from a look into possibilities for accidents. To begin with we have four basic types of accidents:

a) loss of coolant flow
b) accidental reactivity ramp insertions
c) major Na pipe breaks
d) plugging of the central (or an inner) subassembly.

a) The loss of coolant flow accidents happens if all four pumps fail. This is the case if the regular power supply and that of the emergency drive fails. If in addition, as a third coincidence, the safety system fails, the analysis indicates that one has a time intervall of 8o sec, before Na-boiling appears. After this time intervall a Na-voiding accident is to be expected. We come to this point later. One can argue about the probability of a simultaneous mechanical failure. In that event only one further conditioned event, namely that of the failure of the safety system has to happen. But counting the simultaneous mechanical failure of all four pumps and in particular that of the safety system is a picture which is too rough, because these events are in themselves multi-conditioned. In agreement with other assumptions conc. the mechanical design (see chapter 4.) we assume that there will be no such simultaneous mechanical failure. This leads to the conclusion that in the two conditioned case of pump failure (regular and emergency drive failure) the safety system has to act within 8o sec.

b) Accidental reactivity insertions can happen in the beginning of a sequence of events or as the consequence of Na-voiding or core compaction. Na-voiding and core compaction lead to high values of the associated ramp rates (1o - 1oo $/sec) whereas the reactivity insertions in the beginning of a sequence of events lead to low values of ramp rates. For example the loading accident for a central subassembly gives 0.7 $ in 35o msec, that is 2 $/sec.

Or the Na-voiding of the central subassembly leads to a change of 0.o7 $ in, say, 1oo msec, resulting in a ramp rate of 0.7 $/sec. The size of an accidental reactivity insertion by a control rod run away depends on the particular design, but will be always at ramp rates below 1 $/sec.

In the analysis of reactivity insertions the class of ramps between 0 and 2o $/sec was considered. It is useful to make two distinctions. If a reactivity input is below 1 $ the resulting excursion is reactivity value limited, it is largely the total reactivity value and not so much the ramp rate which determines the size of the excursion. If the reactivity is larger than 1 $ it is in the region beyond 1 $ the ramp rate and not the final reactivity value, which determines the size of the excursion. The other distinction is that between low ramp rates leading to Na-boiling before fuel starts melting or vaporization and ramp rates leading to fuel melting or vaporization before Na starts boiling. The limiting ramp rate for this disctinction is a function of the reactor properties, in particular the Doppler coefficient and the average fuel temperature. In certain cases instead of Na-boiling cladding melting can be the relevant criterion.

The reactivity insertion, which the reactor can accept without action of the safety system is 0.25 $. These are rather weak insertions. For all other insertions stronger than that (reactivity value and ramp rate) the scram by a safety system is required. Therefore the logical question is now: What kind of a safety system is required. The detailed studies of Smidt et alii [3] give the answer for the Na-1-design in form of a parametric study. The result is that for a delay time in the safety system of 5o msec and a realistic shut down reactivity versus time characteristic having 15o $/sec ramp rate in its linear part and having a Doppler coefficient of $6 \cdot 10^{-6}$ at 14oo ^{o}K results in a permitted accidental reactivity ramp of 1o $/sec if the beginning of melting is the limit. The value of 15o $/sec assumes, of course, that all shut off rods are in action (1o rods at 1.5 $ each). A delay time of 5o msec is a rather high value and means that more or less conventional shut off systems can be applied. If the delay time would be 1o msec, instead of 1o $/sec about 14 $/sec could be handled. The influence of the Doppler coefficient is limited here. The reason is, that its action reduces the net reactivity to values close to one dollar if the ramp insertion has led to reactivity values larger than one dollar. Before prompt criticality is reached, the Doppler is of very small influence because the large periods of the power level increase do not change the fuel temperature during that period. If $A = T \frac{dk}{dT}$ is the Doppler constant, ΔT the permissible fuel temperature intervall in the case of an excursion and T_o the fuel temperature at normal operation and r_s the accidental ramp rate, the time t_s to fuel melting or vaporization has the characteristic:

$$t_s \sim \frac{1 + A \cdot \ln\left(1 + \frac{\Delta T}{T_o}\right)}{r_s}$$

Only if $A \cdot \ln\left(1 + \frac{\Delta T}{T_o}\right)$ gives a significant contribution compared to one the Doppler has a stronger influence on the characteristics of the safety system. That is the case for carbide fuel at low rod power. At 5o msec delay time, a Doppler coefficient of $4 \cdot 1o^{-6}$ leads to 9.3 \$/sec, a Doppler coefficient of $8.5 \cdot 1o^{-6}$ to only 1o.5 \$/sec permissible accidental ramp rate. The function of the Doppler coefficient is therefore to serve as an inherent safety mechanism against supercritical fast reactivity insertions and particularly to guarantee stability. The impact of the incertainties in the nuclear calculation of the Doppler coefficient on the design of the safety system is limited.

One important conclusion to be drawn from these arguments is that an inconventional safety system with very short delay times (1 msec or so) is required only in the case of accidental reactivity ramps much larger than, say, 2o \$/sec. But it is difficult to think of such accidental ramp rates in the beginning of a sequence of events. Only Na-voiding of the core could give larger values, but here in this line of arguments Na-voiding is only a consequence of accidents which are accompanied by a failure of the safety system. We therefore conclude that for the safety system a conventional version of it is sufficient. This differs from the earlier Karlsruhe thinking.

Whether the event of an accidental ramp reactivity insertion is simple conditioned or two conditioned can be subject for a more detailed safety discussion but it is of no importance here in this context.

c) Major Na-pipe breaks between the reactor vessel and the intermediate heat exchanger interrupt the circuit of natural convection for a primary circuit of the loop type. Such a break would require shut down and emergency cooling in the vessel.

According to the arguments of chapter 4. such a major break of a pipe is excluded in the Na-1-study by definition. The forthcoming Na-2-study might eventually take up this point. The reactor vessel itself has a double wall and therefore a break is definitely excluded.

d) Before going to the accident of a major Na-voiding of the core, the accident of the plugging in the central subassembly shall be considered. If such a plugging takes place the response of the neutron flux might be very small due to the low reactivity value of Na-voiding of the central subassembly. No reaction of the safety system therefore takes place and the Na of the central subassembly might start boiling. Smidt has emphasized [3], that e.g. in the case of the Na-1-design at 28 °/o of normal flow, boiling of the Na occurs, but already at 22 °/o of the normal flow unstable flow conditions may occur thus keeping the coolant in the channel. This leads to the beginning of a more serious sequence of accidental events which will be discussed in the next chapter. One can argue that boiling can perhaps be detected acustically and therefore be used as an initiation of a scram signal. As long as the safety system works this kind of an accident might be prevented therefore.

But it is also possible that the sodium superheats and does not boil [3]. This can possibly not be detected by an instrumentation. Thereafter sudden desuperheating may well lead to destruction of the central subassembly and its neighbours thus giving a major core damage ending up in a serious core explosion. This will be discussed in the next chapter.

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This difficulty of an undetectable plugging and superheating accident is considered to be an open point and so far an unresolved problem in the realm of the assumption of an uneffected safety system.

In order to attack this problem much more has to be known on the mechanism of two phase flow and superheat both theoretically and experimentally. The paper of Fischer and Häfele [10] and that of Friz [11] are considered to be a first step in this direction. Along the same lines one also has to fear a break of the cladding at the fission gas plenum. This also might lead to a propagation and result in a plugging of the coolant channel. This danger is somewhat emphasized in the Na-1-design where the fission gas plenum is at the lower end, but putting it on the top of the fuel pin probably does not change the situation drastically.

We realize that all considerations of this chapter with the exception of the plugging and superheating accident do not lead to any larger accident in case the safety system works, in particular we realize that the size of the Na-void effect did not influence our considerations. The mechanism of the plugging and superheating accident requires therefore further thorough investigations. But the accidents of the type a) - d) lead to a major damage in case of a simultaneous failure of the safety system. The sequence of events leading to such a major damage all involve the Na-voiding of major parts of the core. This leads to the maximum hypothetical accident.

7. The Maximum Hypothetical Accident (MHA)

We start with the assumption of a reactivity ramp insertion as it was discussed under b) in the last chapter. The accidents discussed there are of such a low value, that sodium boiling starts before fuel melting ends and vaporization of the fuel starts. In the case of the Na-1-design the limit for these reactivity ramps is 3 ¢/sec. We also assume that the safety system fails to react. Now it is not sufficient to assume that the boiling begins at the upper end of the core where the sodium void effect is negative. According to the argument of Smidt [3] we have to assume unstable flow conditions and in this case voiding starts at the center of the core at the place of the largest positive void effect. If one wants to evaluate the ramp rate associated with Na-voiding, the situation is as follows:

The original calculation of the maximum integral void effect in the Na-1-reference design gives a maximum value of 3 ¢ for sodium voiding, the latest results indicate a higher value close to 7 ¢. In the analysis of this accident the uncertainty of the nuclear calculation of the void effect appears for the first time to be of relevance. If one wants to continue to evaluate the maximum ramp rate a second uncertainty appears, namely that of the time scale of throwing out the sodium. Here the full scope of the Na two phase flow appears. As Fischer and Häfele show in their paper [10] for low amounts of gas in the two phase mixture one has to expect a rather low value of the velocity of sound (easily as low as 10 m/sec). This means that all kind of shock wave complications have to be expected in the pattern of flow. The time scale is probably uncertain to a much higher degree than the reactivity value of voiding. Both uncertainties combine if the ramp rate is calculated. Presently we bridge this difficulty by simply assuming a ramp rate of 100 ¢/sec.

Following the original initiating excursion one finds, that together with the boiling of Na the fuel melts. The FORE analysis indicates that the hotter part of the fuel is already molten when the Na starts boiling. If the molten fuel touches the can the heat resistance in the gap disappears and the temperature of the cladding is suddenly increased for about 500°C. The immediately following formation of bubbles of Na vapor is consistent with the above mentioned second uncertainty of the time scale for Na-voiding. This formation of bubbles of Na vapor means that there might be a Na steam explosion of the BORAX type.

After the melting of the can fuel compaction inside and outside of the original fuel element could start. This would lead to another reactivity ramp. Both ramps, that of the Na-voiding and that of fuel compaction, would add if the two involved mechanical processes would not interfere. They would lead to a nuclear excursion and subsequent explosion. Superimposed is the Na-steam BORAX type explosion.

We have not yet a clear idea on the patterns of the interactions of these two processes. Here is another open problem of great importance.

It is useful to realize the difference between this kind of an analysis and that for a small reactor with metallic fuel elements like EFFBR or EBR II. There we have a negative void effect. Any displacement of somewhat larger amounts of Na shuts the reactor down, the two ramps involved there subtract and don't add as above. The metal fuel melts easily and drops to the bottom of the core and may form a second critical mass down there. This second critical mass is not at the place of the first critical mass, the core. The excursion in this second critical mass is known as second excursion which constitutes the MHA (or possibly maximum credible accident). In our case the place for the accidential excursion is more at the place of the original core and the BORAX type Na-steam explosion is superimposed; the ramps add up and there is no time for a quiet melt down.

The plugging and superheating accident for the central subassembly can develop to a similar final picture as described above. Sudden desuperheat destroys the central subassembly and its neighbours. The probable interruption of the coolant flow there leads to Na-boiling (or Na superheat, which in turn would increase the area of subassembly destruction even further by a repetition of this mechanism). The ejection of Na of a somewhat greater number of subassemblies leads now to a reactivity ramp of a relevant height and could initiate the scram rods. But here we have to assume that even if the safety system as such works the safety rods don't drop into the core because of mechanical distortions there. Now the sequence of events develops similar to the above described picture.

In spite of all this the amount of kinetic energy build up in an excursion of a given ramp rate was calculated by Ott, Böhm /¯12_7 and later by Küsters. They used a Bethe-Tait approach using a spherical model for the supercritical mass. It is realized that extremely flat cores of, say, $\frac{H}{D} = \frac{1}{6}$ may better use a slab model which should give smaller energy releases. This will be done in the forthcoming months.

Following the Bethe-Tait type calculation another uncertainty is to realize. This is the uncertainty of the equation of state for a realistic fuel, that is a mixture of UO_2-PuO_2 and fission products containing fission gases to a various degree.

At Karlsruhe originally an equation derived from the measurements of Ackermann /¯13_7 were used. This led to a rather quick pressure build up and limited the energy release nicely. But then we detected the large difference against the energy release obtained by the use of the Wolfe-Meyer /¯14_7 equation of state. To have a safe approach we now use only this equation of state. The Doppler coefficient is included in the calculation. This results in a remarkable reduction of the energy release. The following table gives the total energy release and that of kinetic energy for different values of the Doppler coefficient at 2500^{o}K, the initial flux and two ramp rates.

Table 4

α_R = 500 ¢/sec.

DC	Φ_o	E /kg TNT/
$-8 \cdot 10^{-6}$	10^{10}	256
	10^{14}	48
	10^{18}	114
$-4 \cdot 10^{-6}$	10^{10}	539
	10^{14}	84
	10^{18}	280
$-1 \cdot 10^{-6}$	10^{10}	5 474
	10^{14}	2 848
	10^{18}	882
0	10^{10}	29 814
	10^{14}	18 112
	10^{18}	8 830

Table 5

α_R = 100 ¢/sec.

DC	Φ_0	E [kg TNT]
$-8 \cdot 10^{-6}$	10^{10}	34
	10^{14}	7.4
	10^{18}	17
$-4 \cdot 10^{-6}$	10^{10}	79
	10^{14}	85
	10^{18}	39
$-1 \cdot 10^{-6}$	10^{10}	43
	10^{14}	284
	10^{18}	239
0	10^{10}	6 019
	10^{14}	4 196
	10^{18}	2 055

The Na-1-reference design has a Doppler coefficient between 6 and 8 · 10^{-6} at 1400°K. At 2500°K this leads to a value of 3.3 and 4.5 · 10^{-6} respectively. The correct value of the Doppler coefficient and the initial flux is a principal difficulty in calculating the energy release.

Assuming a value of 4.10^{-6} for the Doppler gives 85 kg of TNT for 100 ¢/sec and $\Phi_o = 10^{14} \frac{n}{cm^2 sec}$, at 500 ¢/sec we have for $\Phi_o = 10^{10} \frac{n}{cm^2 sec}$ 539 kg instead.

A major experimental effort is in preparation at Karlsruhe in order to investigate the equation of state experimentally.

8. The Design Basis Accident (DBA) and the Design of the Containment

About a year ago when the Karlsruhe reference design, Na-1, neared its completion the uncertainty of calculating the energy release in case of a MHA was larger than it appears today. So it was decided to accept in case of the Na-1-design the philosophy that the safety system does not fail. Reporting on the results of the third Geneva Conference W. Häfele and P. Engelmann pointed out that this philosophy is accepted by many fast reactor groups /‾15_7. The mechanism of the plugging and superheating mechanism was not yet detected. This excludes the MHA as a DBA. The DBA is therefore a large sodium fire. The steel container is designed to withstand this maximum pressure of 2.5 atü. But it is highly unprobable that this pressure builds up because only a fraction of the sodium can probably go to fire. Namely all the compartments containing the primary circuit have a nitrogen atmosphere and only the case of a major explosion, which is excluded here, could lead to such a fire of all the sodium. The diameter of the containment building is only 28 m. It results from the size of the primary circuit. Its height is determined by the installation and maintenance or repair procedures of the components of the primary circuit. At the inside of the steel containment a layer of concrete protects against temperature of a fire and mechanical damages.

The Na-1-reference design shows already some neighbourhood to the two containment solution of the SEFOR reactor. The DBA there is a nuclear explosion of 90 kg TNT. It will be contained in the compartments of the concrete structure of the primary loop. The second container, the steel shell, in the case of SEFOR, shall withstand the pressure of an overall sodium fire. Therefore it will not be too difficult to extend the Na-1-reference design to a DBA of the SEFOR type. To exclude the MHA as a DBA is therefore a less severe assumption. In the forthcoming weeks an effort will start in order to evaluate the capital cost potential of such change of philosophy. Thereafter the basis for the DBA of the Na-2-design will be selected.

9. Capital Costs of the Na-1-Reference Design, its Areas of Greatest Potential for a Reduction and its Competition with Breeding and Safety.

One of the objectives of the Na-1-reference design was to get a rough idea on the costs and its distribution in order to use this information as a guidance for further research and development work.

The fuel costs are 0.3 DPf/kWh = 0.75 $ mills/kWh at a Pu price of 10 $/g. This includes interest for inventory, refabrication, reprocessing and the first core. It also includes the sale of bred material. The capital costs are 462.6 Mio DM = 116.0 Mio $ or 116 $/kWe. The 462.6 Mio DM are the sum of 262.6 Mio DM for the conventional part = 58 % of the total capital costs. 200.0 Mio DM are assumed to be the costs for the reactor part, that is 42 % of the total sum. But the actual capital cost evaluation led to a value of only 116.95 Mio DM, 83.05 Mio DM are considered to be a "contingency" due to general uncertainties in the area of costs for large Na components. If this "contingency" is excluded, the total sum is only 380.0 Mio DM = 95 $/kWe with 68 % for the conventional part and only 32 % for the reactor part.

The amount of 116.95 Mio DM = 29 Mio $ = 100 % shall now be the reference value for comparisons. 11 Mio $ = 38 % is for the pumps, all the devices of the cooling circuits provide 19 Mio $ = 65 % of the total costs of the reactor part. The reactor itself gives only 5 Mio $, that is 17 %.

The reactor building should be in the order of 6 Mio $.

As being reported in the paper of Smidt /‾3_7 also the change from oxide to carbide fuel was considered. There are two ways to use the better properties of the carbides: It is possible to use the same rod power and to enjoy the higher breeding ratio, both internal and external and the lower average temperature only, or to operate the reactor at higher rod power additionally. Smidt's analysis indicates, that in the second case an accordingly higher cooling flux of sodium is required. This leads to higher pressure drops in the primary circuit and therefore to more expensive pumps and more structural material for the subassemblies, which in turn lowers the breeding ratio. The higher cost for the pumps nearly outbalance the gain in fuel cycle cost, if they are capitalized over a period of 15 years at 9.7 %. The two choices to use the carbides instead of the oxides result in nearly the same cost reduction of about 20 Mio DM = 5 Mio $.

We already saw, that the components of the coolant circuits are the area with the largest potential for possible cost reduction. In the Na-1-study we used cheap ferritic steel whereever it was possible therefore. Up to now the cheap SS of the 304 or 316 type are forbidden for high temperature use. If, however, the use of these cheap SS were permitted, the thermal efficiency could be increased by perhaps 5 %. This could lead to a decrease of 0.2 $\frac{\text{mills}}{\text{kWh}}$ at 80 % load factor. One should realize here, that already a decrease of about 0.3 $\frac{\text{mills}}{\text{kWh}}$ is the incentive for the light water superheat development. The penalty for the change from the ferritic to the cheap SS would amount to only 0.03 $\frac{\text{mills}}{\text{kWh}}$. If on the other hand we continue to be forced to use a X8 Cr Ni Nb 16 13 type of SS gain and loss of costs may just cancel each other. The optimization of efficiency - capital costs and fuel cycle costs requires further attention.

If one tries to evaluate this cost situations one arrives at the following conclusions:

a) the carbides have a cost advantage over 5 % with respect to the assuming that the fuel cycle cost are capitalized over 15 years at 9.7 %. This advantage is large enough to be interesting, but it is not so large that the Karlsruhe group would follow a parallel development of both, the oxides and the carbides. The development of the carbide fuel is a meaningful second step. It also gives higher breeding and in case of the same rod power as in the case of the oxides the advantage of a higher inherent safety. Cost advantage, safety and breeding all tend in the same direction.

b) Among the more direct reactor costs it is definitely the part of the Na-components which has the largest cost potential. In particular the pumps are very expensive and for pumps a well targeted development program is advisable. But also the questions of materials of the Na-circuits and -components are of interest, especially on the background of present engineering regulatory rules in Germany. The situation there entrances a serious consideration of the use of cheap ferritic steel in competition with the question of thermal efficiency.

c) The reactor building of the Na-1 is a 6 % item of the whole plant. A more complicated building on the two container principle being able to withstand explosion of, say, 100 - 200 kg TNT is subject of further investigation. It is estimated to increase the cost from 6 to 7 or 8 % of the total costs. Therefore this does not result in a heavy competition between safety and economics.

d) The reactor itself is a 5 % item. Therefore any complication there which appears to be necessary for safety or breeding again does not afflict heavily the economics.

Generally one can say that the often cited competition between economy, safety and breeding does exist, but is not too heavy. In the case of fuel <u>all</u> three aspects are <u>favourably</u> afflicted by the use of the carbides.

10. Conclusions

1) A 1000 MWe fast breeder reactor with Na cooling appears technically feasible.

2) The safety properties of such a reactor do not seem to constitute a major safety problem. Soft accidential reactivity insertions as defined in chapter 6, that is below 0.25 $, can be accepted in the Na-1-reference design without the action of the safety system. More severe reactivity insertions require the use of a rather conventional safety system (about 50 m sec delay time before mechanical movement starts).

3) The Doppler coefficient allows for the use of such a rather conventional safety system und guarantees over all stability. But its actual size does not have too large an impact on these design characteristics. In case the safety system works the size of the Na-void effect does not have a larger feed back on the design.

4) Within the realm of a working safety system the recently detected plugging and superheating accident of a central subassembly requires further attention.

5) If the safety system can fail major accidents are possible. Here the size of the Na-void effect is of great importance. Its positive sign leads to an addition of the ramp rates due to Na-voiding and fuel compaction. Both, the nuclear excursion and a Na-steam explosion superimpose. The Doppler coefficient decreases remarkably the release of kinetic energy.

6) Work should be done in the area of:
   a) the nuclear evaluation of the Na-void effect
   b) two phase flow of Na
   c) Na-steam explosion and the mechanism of core destruction.
   d) the equation of state of a realistic oxide fuel

7) The expected fuel cycle cost for a Na-1-reference design is at 0.7 $\frac{\text{mills}}{\text{kWh}}$. The capital costs are at $\frac{100\ \$}{\text{kWh}}$, the breeding ratio is at. 1.4, the doubling time (of a population of fast breeder)

is about 7 years. This is very satisfactory [16].

8) The choice of fuel is oxide. Carbide could lead to a decrease of the order of 5 % of the total costs (over 15 years). The development of a carbide fuel is therefore meaningful as a second fuel development after the oxide fuel. Carbide fuel cannot be used in a steam cooled design.

9) The components of the cooling circuits make up nearly 65 % of the cost of the more direct reactor part. The pumps alone give 38 %. Careful development of cheap large Na components is in that sense important. The cost optimum between Na outlet temperature, that is thermal efficiency, and capital costs requires further attention.

10) The principle of large fast breeder reactors is to have fuel cycle costs below 1 $\frac{\text{mill}}{\text{kWh}}$. Therefore large scale fuel testing up to a burn up of 100 000 $\frac{\text{MWd}}{\text{to}}$ as a problem of long testing times is of great importance.

APPENDIX

Data of the Karlsruhe Reference Design Na-1-

1. General Data

Reactor Type	Heterogeneous Fast Breeder	
Reactor Power Thermal	(MW_{th})	2 500
Electrical	(MWe)	1 000
Fuel	UO_2-PuO_2 87 % theoret. Density	
Fertile Material	UO_2, U-Metal	
Coolant Primary		Na
Secondary		Na
Tertiary		H_2O
Total Breeding Ratio	(-)	1,385
Doubling Time (1/3 out of Pile Inventory) - for a Population of Fast Breeders	(a)	7,2

2. Reactor

2.1 Reactor Core

2.1.1 Construction and Composition

Form	Upright Zylinder	
Total Diameter	(mm)	3 766
Total Height	(mm)	1 755
Fuel Region Diameter	(mm)	2 860
Height	(mm)	955

H/D Ratio	(-)	1/3
Thickness of Blanket Radial	(mm)	400
Axial	(mm)	400
Total Number of Zones in Radial Direction (numbered from the Center outward)		4
in Fuel Region	(-)	2
in Radial Blanket Region	(-)	2
Total Number of Cells	(-)	397
in Fuel Region	(-)	229
in Radial Blanket Region	(-)	168
Form of Cell	Hexagonal	
Width over Flats	(mm)	178,5
Overall Length of Subassembly	(mm)	3 790
Weight of Fuel Subassembly	(kp)	315
Weight of UO_2-Blanket Subassembly	(kp)	427
Weight of U-Metal Blanket Subassembly	(kp)	587
Thickness of Subassembly Cladding	(mm)	4
Diameter of Fuel Pin	(mm)	6,7
Diameter of Blanket Pin	(mm)	11,3
Cladding Thickness of Fuel Pin	(mm)	0,35
Volume Ratio Cladding/Fuel	(-)	0,25
Cladding Thickness of Blanket Pin	(mm)	0,6
Volume Ratio Cladding/Fertile Material	(-)	0,25
Cladding Material	Incoloy 800	

Number of Fuel Pins per Sub-assembly		(-)	331
Number of Blanket Pins per Sub-assembly		(-)	169
Fission Gas Plenum (below)		(mm)	800
Max. Fission Gas Pressure in Fuel Pin		(kp/cm^2)	70

Fuel Region		Zone 1	Zone 2
Outer Diameter	(mm)	2 053	2 860
Number of Cells	(-)	118	111
Volume Fraction			
Coolant	(v/o)	50,0	50,0
Structural Materials	(v/o)	19,6	19,6
Fissile Material	(v/o)	3,2	4,1
Fertile Material	(v/o)	27,2	26,3
Number of Fuel Subassemblies	(-)	105	105
Number of Control Rod Sub-assemblies	(-)	12	6
Fraction of Total Coolant per Fuel Cells	(-)	0,465	0,465
Fraction of Structural Material per Fuel Cell	(-)	0,121	0,121

Radial Blanket Region		Zone 3	Zone 4
Outer Diameter	(mm)	3 213	3 766
Number of Cells	(-)	60	108
Number of Blanket Sub-assemblies	(-)	60	108
Volume Fraction			
Coolant	(v/o)	29,8	29,8

Structural Materials	(v/o)	21,9	21,9
Fertile Material	(v/o)	48,3	48,3

2.1.2 Thermodynamic Data

Thermal Power Core	(MW_{th})	2 410
Blanket Axial	(MW_{th})	48
Blanket Radial	(MW_{th})	42
Coolant Inlet Temperature	(°C)	430
Max. Outlet Temperature	(°C)	630
Avg. Outlet Temperature	(°C)	580
Max. Coolant Velocity	(m/sec)	6,6
Pressure Loss		
in Core	(kp/cm^2)	1,25
in Axial Blanket	(kp/cm^2)	1,05
in Fission Gas Plenum	(kp/cm^2)	1,05
Max. Fuel Temperature	(°C)	2 412
Max. Heat Flux	(W/cm^2)	334
Hot Channel Factors		
Coolant Heating	(-)	1,33
Heat Transfer Coefficient	(-)	1,25
Rod Power	(-)	1,24
Hydraulic Diameter in Fuel Region	(mm)	6,8
Ratio of Avg. to Max. Power Radial	(-)	0,82
Axial	(-)	0,80

		Zone 1	Zone 2
Max. Rod Power	(W/cm)	566	539
Ratio of Avg. to Max. Power			
Radial	(-)	0,897	0,745
Axial	(-)	0,80	0,80

Constants

Max. Heat Transfer Coefficient between Cladding and Coolant	($W/cm^2 \cdot {}^oC$)	8,4
Thermal Conductivity of Cladding (600°c)	($W/cm \cdot {}^oC$)	0,21
Heat Transfer Coefficient between Fuel and Cladding	($W/cm^2 \cdot {}^oC$)	0,75
Thermal Conductivity of Fuel	($W/cm \cdot {}^oC$)	0,032

2.1.3 Physics Data

Critical Mass (Pu-239 + Pu-241)	(kg)	2 015
Total Neutron Flux at Core Center	($cm^{-2} \cdot sec^{-1}$)	10^{16}
Effective Neutron Lifetime	(sec)	$3{,}38 \cdot 10^{-7}$
Effective Fraction of Delayed Neutrons	(%)	0,3548
	($)	1,0
Worth of a Fuel Subassembly at Core Center $\frac{\Delta K}{K}$	(-)	$2{,}49 \cdot 10^{-3}$
	($)	0,7
Doppler Constant $A_{Dopp} = T \cdot (\frac{1}{K} \cdot \frac{dK}{dT})$	(-)	$-\ 1{,}19 \cdot 10^{-2}$
Doppler Coefficient $\frac{1}{K} \cdot \frac{dK}{dT}$	(${}^oK^{-1}$)	$-\ 8{,}5 \cdot 10^{-6}$
Greatest Possible Reactivity Worth due to Partial Loss of Na $\frac{\Delta K}{K}$	(-)	0,01108
	($)	3,06
Na Temperature Coefficient	(${}^oC^{-1}$)	$3{,}925 \cdot 10^{-6}$
Breeding Ratio Zone 1	(-)	0,556
Zone 2	(-)	0,334

Axial Blanket		(-)	0,254
Radial Blanket		(-)	0,241
Total		(-)	1,385

		Zone 1	Zone 2
Fuel Composition:			
Pu-239	(a/o)	9,8726	12,9028
Pu-240	(a/o)	4,7013	6,1442
Pu-241	(a/o)	0,7835	1,0241
Pu-242	(a/o)	0,3134	0,4096
U-238	(a/o)	84,3292	79,5193
Enrichment Fissionable Pu/(Pu+U)	(a/o)	10,6561	13,9269
Atom Ratio Fertile/Fissile Material	(-)	8,3843	6,1803
Volume Ratio Fertile/Fissile Material	(-)	8,7110	6,4083

Composition Blanket Material	U-235	(a/o)	0,7
	U-238	(a/o)	99,3
Core Volume		(ltr)	6 132
Rating Max.	(MW/kg/Pu-239+Pu-241/)		2,14
	(KW/cm^3/PuO_2+UO_2/)		2,0
Avg.	(MW/kg/Pu-239+Pu-241/)		1,196
	(KW/cm^3/PuO_2+UO_2/)		1,285
Burn Up Max. (Axial Average)	(MWd/t/U+Pu/)		100 000
Avg.	(MWd/t/U+Pu/)		85 000
Power Density Max.	(MW/ltr.)		0,599
Avg.	(MW/ltr.)		0,393

2.2 Reactor Vessel

Total Height	(mm)	16 900
Outer Diameter	(mm)	7 900
Thickness of Wall	(mm)	30
Core Support Grid Plate		
Diameter	(mm)	4 500
Height	(mm)	1 300
Material	X8 Cr Ni Nb 16 13	
eventually	10 Cr Mo Ni Nb 9 10	
Max. Permissible Temperature of Containment	(^{o}C)	480
Max. Permissible Pressure on Containment	(kp/cm^2)	6

3. Primary Loop

3.1 Pipelines and Pumps

Coolant Volume	(m^3)	780
Activity of Coolant Na 24	(C/cm^3)	$1,36 \cdot 10^{-1}$
Na 22	(C/cm^3)	$2 \cdot 10^{-6}$
Diameter of Pipe from Reactor to Intermediate Heat Exchanger	(mm)	1 700
Coolant Velocity in Pipe	(m/sec)	3,5
Diameter of Coaxial Pipe from Intermediate Heat Exchanger to Reactor inside	(mm)	1 900
outside	(mm)	2 200
Coolant Velocity in Pipe	(m/sec)	7,7
Number of Pumps	(-)	4
Power Input of each Pump	(KW)	2 940

Flow Rate through each Pump	(kg/sec)	3 296
Pressure Loss in Primary Loop	(kp/cm^2)	5,6
Coolant Avg. Circuit Time	(sec)	62,4

3.2 Intermediate Heat Exchanger

Number of Exchangers	(-)	2
Type: Helical Tube Counter Current Heat Exchanger		
Total Height	(mm)	16 000
Outer Diameter	(mm)	4 600
Volume of Coolant Primary	(m^3)	125
Secondary	(m^3)	50
Number of Exchanger Tubes	(-)	4 385
Diameter of Tubes	(mm)	20x1,5
Material	X8 Cr Ni Nb 16 13	
eventually	10 Cr Mo Ni Nb 9 10	
Surface Area per Heat Exchanger	(m^2)	2 792
Avg. Log. Temperature Difference	(^oC)	40
Max. Temperature Difference between Surfaces	(^oC)	70

Primary Side

Coolant Flow Rate per Exchanger	(kg/h)	$23{,}73 \cdot 10^6$
Pressure Loss	(kp/cm^2)	0,43
Coolant Velocity between the Tubes	(m/sec)	1,6
Coolant Inlet Temperature	(^oC)	580
Outlet Temperature	(^oC)	430

Secondary Side

Coolant Flow Rate per Exchanger	(kg/h)	$17{,}75 \cdot 10^6$
Pressure Loss	(kp/cm^2)	2,63
Coolant Velocity in the Tubes	(m/sec)	5,9
Coolant Inlet Temperature	(°C)	360
Outlet Temperature	(°C)	560

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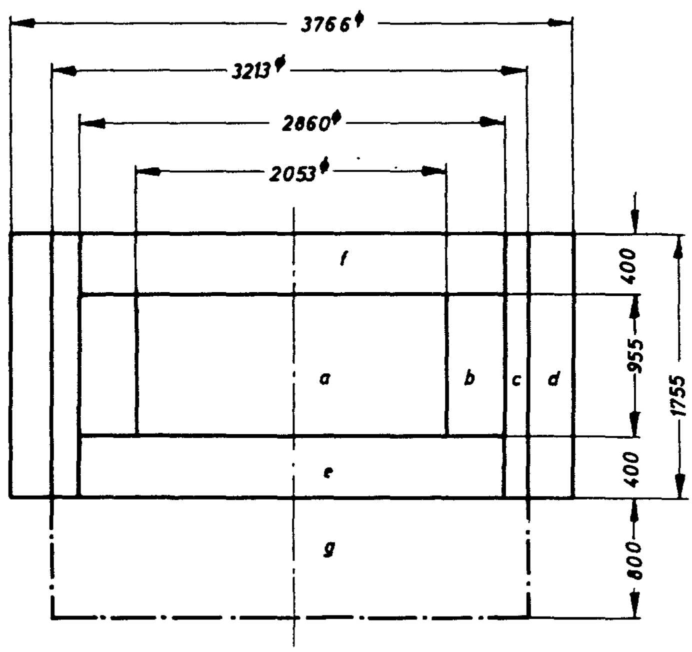
Deutsches Atomforum e.V. and Kernforschungszentrum Karlsruhe, FORATOM-Kongress, Frankfurt, September 1965
Kernforschungszentrum Karlsruhe, KFK-Report Nr. 366



a innere Spaltzone (Zone 1)
b äußere Spaltzone (Zone 2)
c innere Brutzone (Zone 3)
d äußere Brutzone (Zone 4)
e untere axiale Brutzone
f obere axiale Brutzone
g Spaltgasspeicher

Abb. 1 Zonenaufteilung des Reaktorkernes

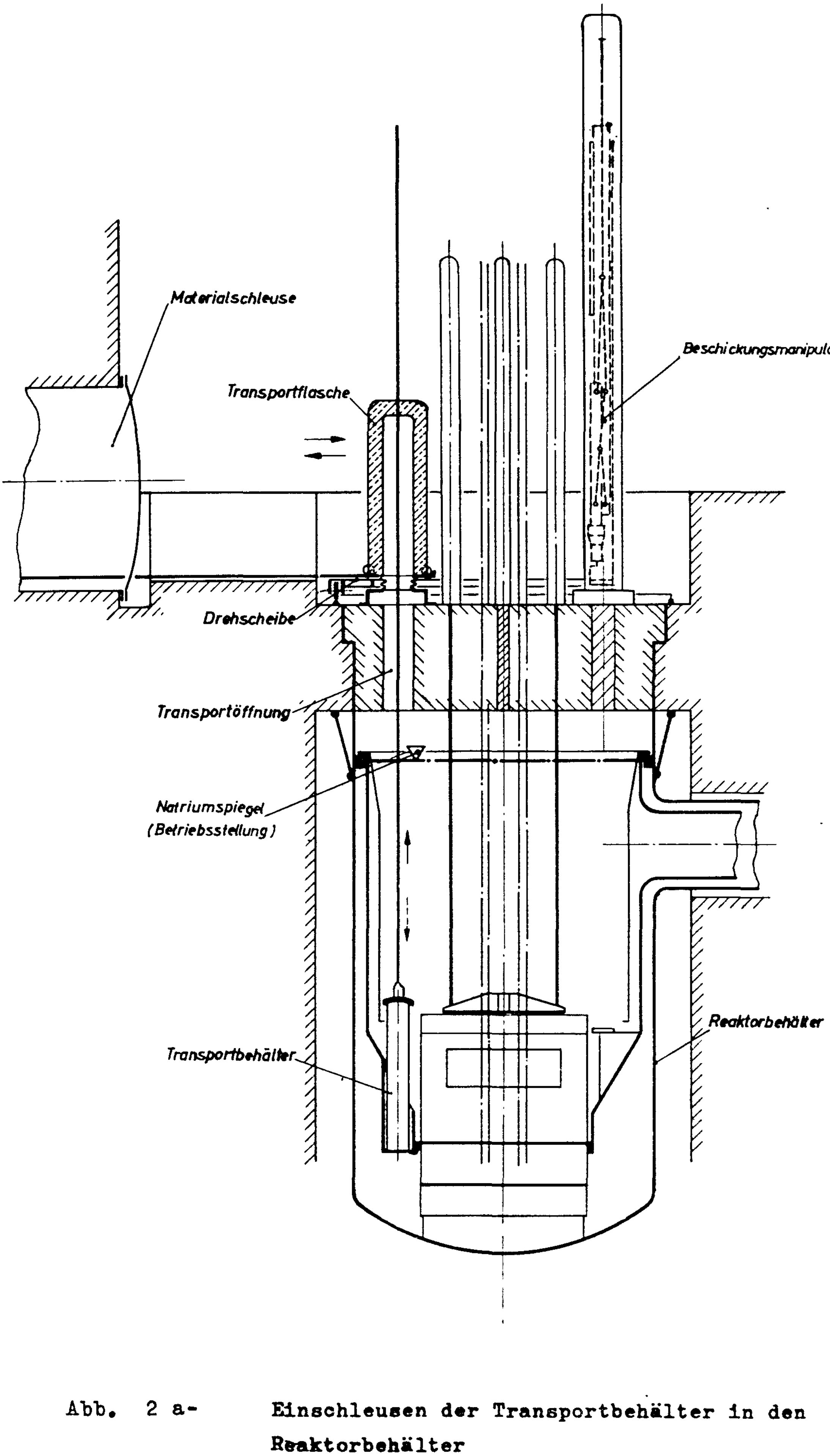


Abb. 2 a- Einschleusen der Transportbehälter in den Reaktorbehälter

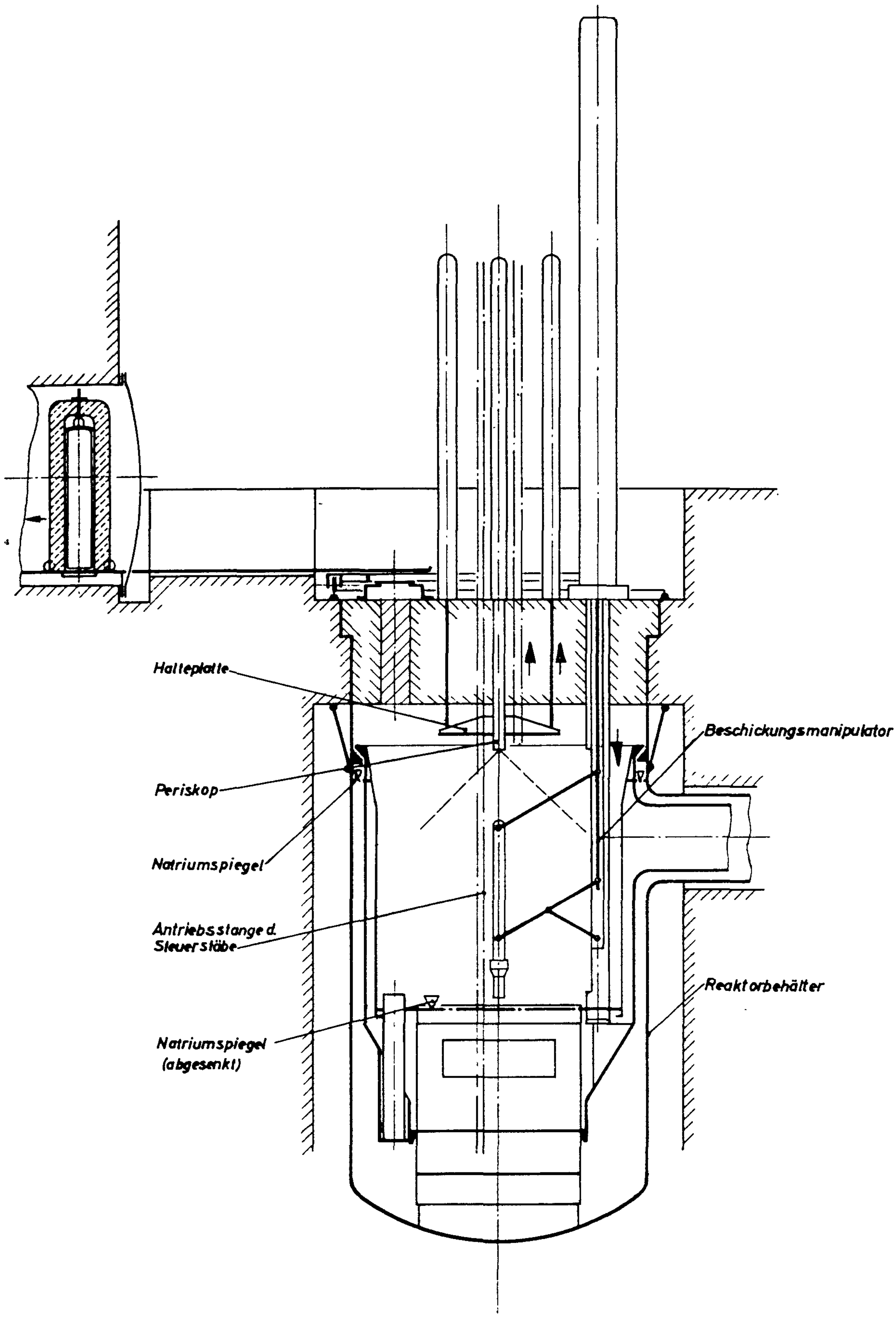


Abb. 2 b Vorbereitung des Reaktors für den Brennstoffwechsel

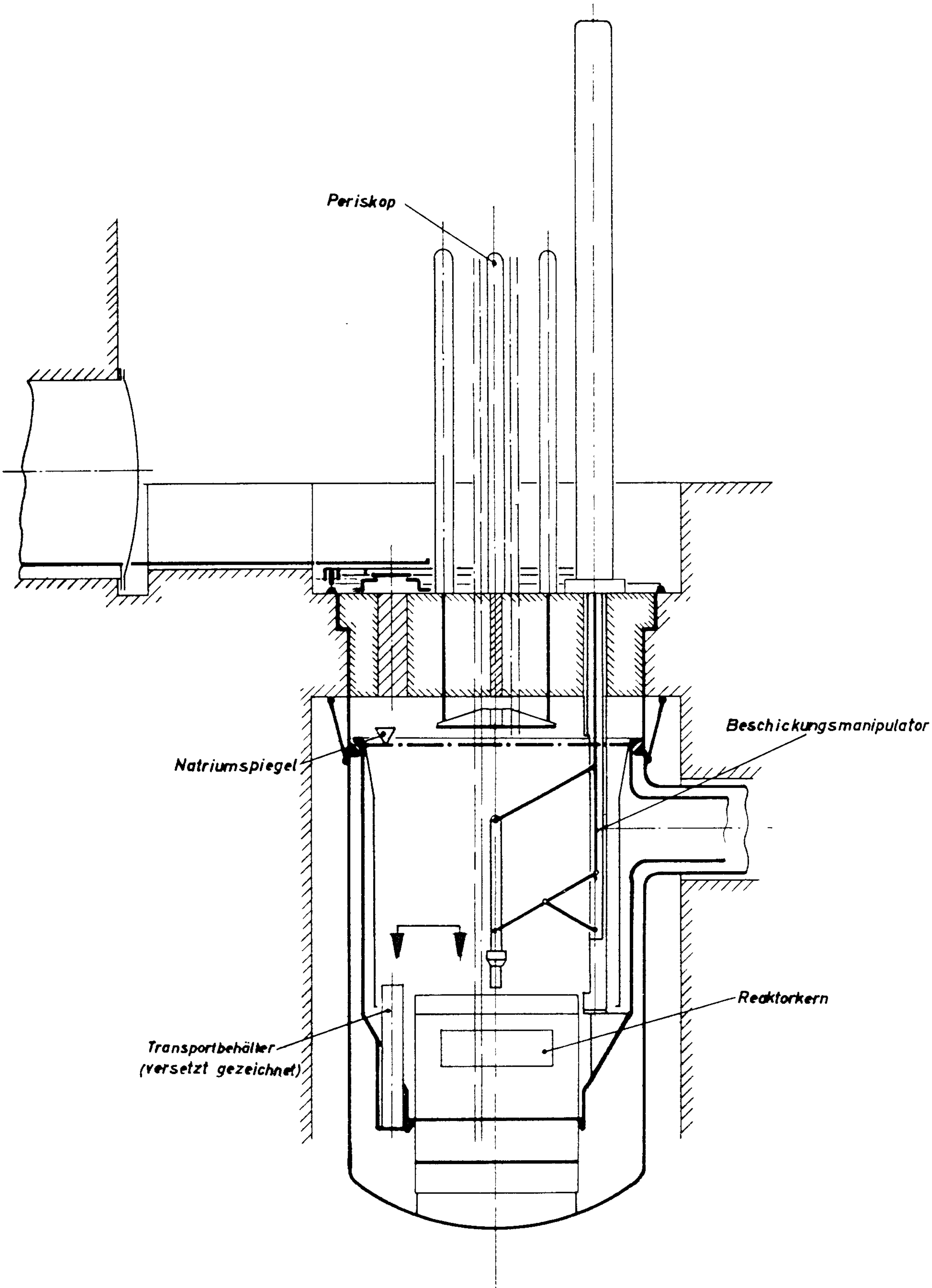
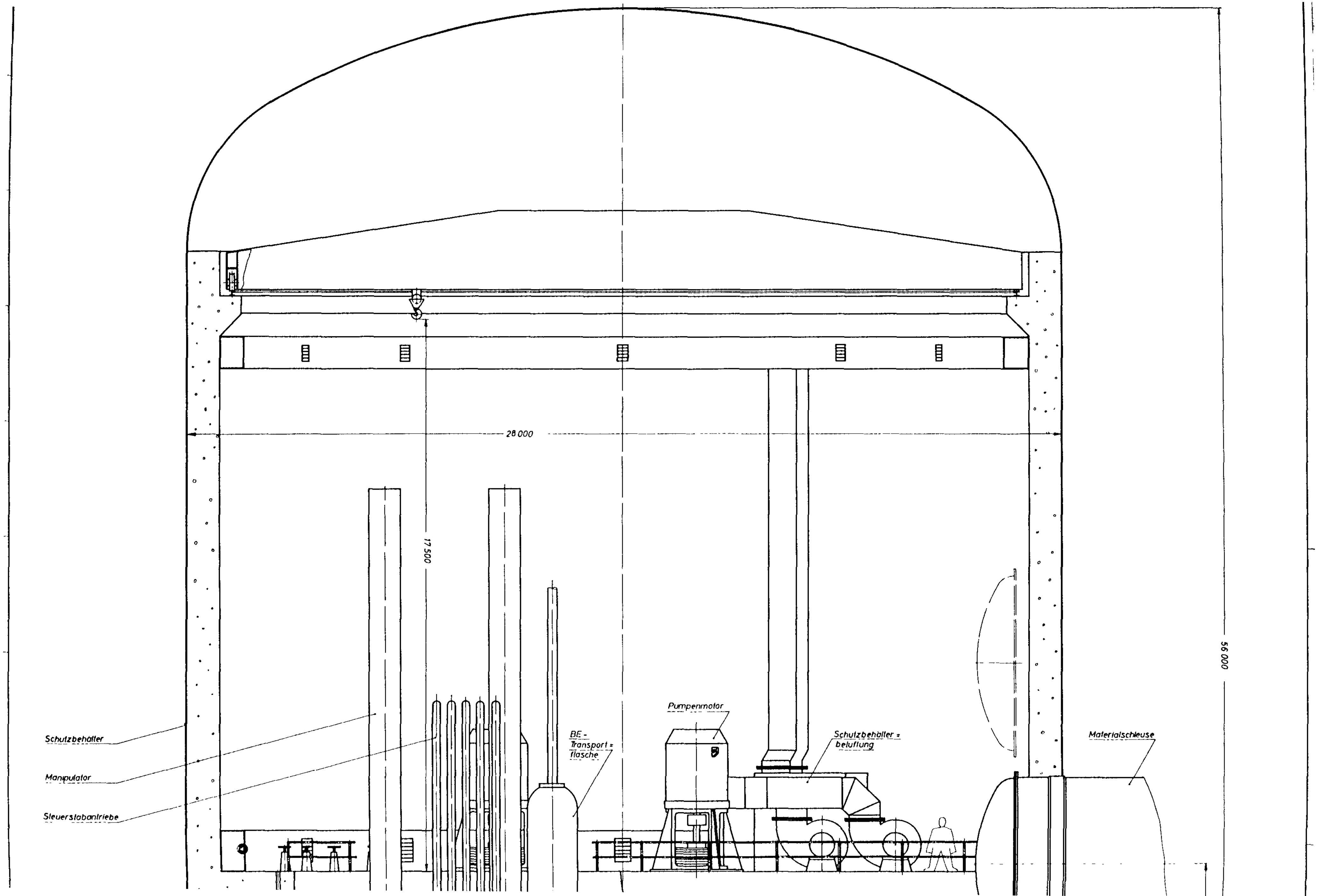
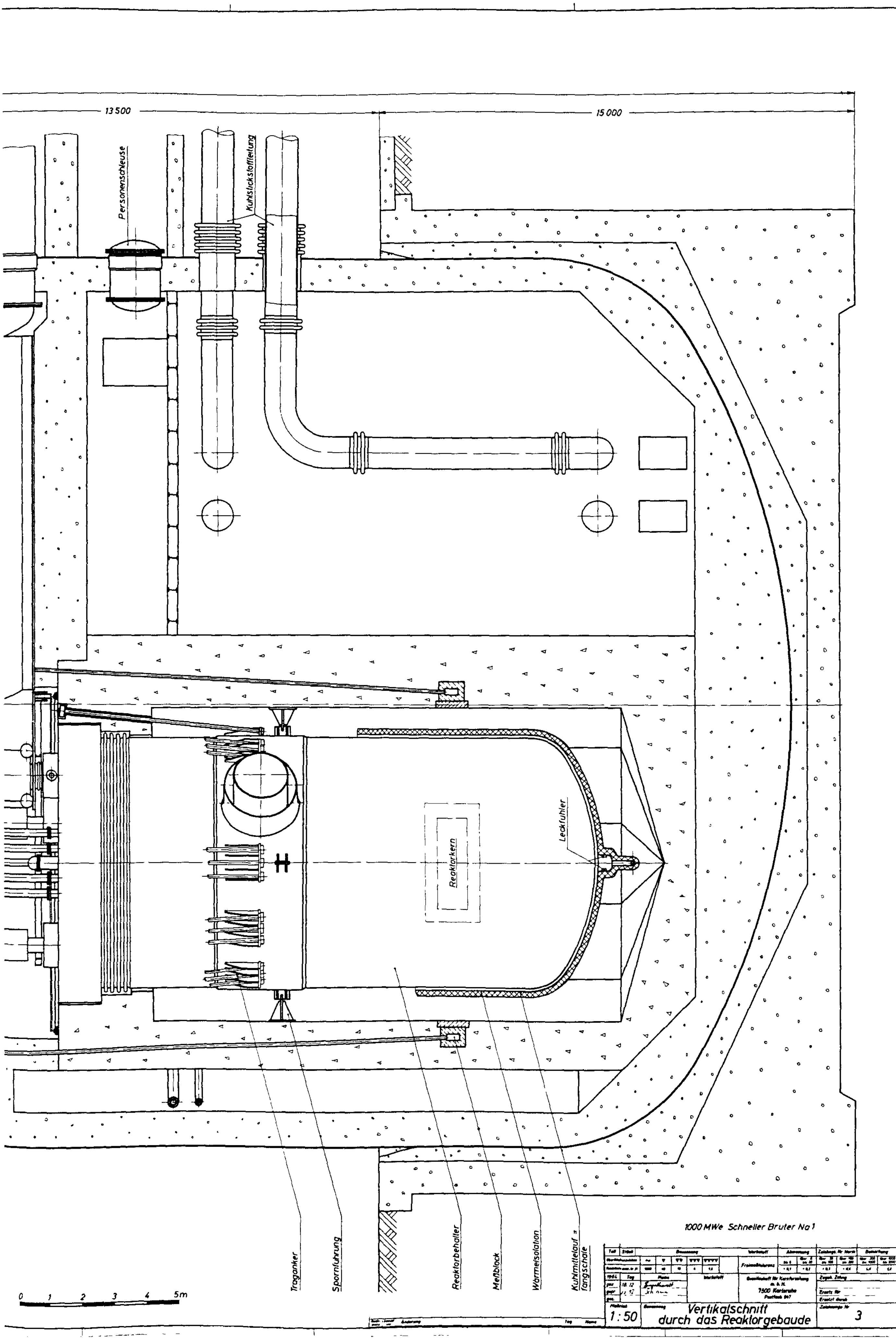
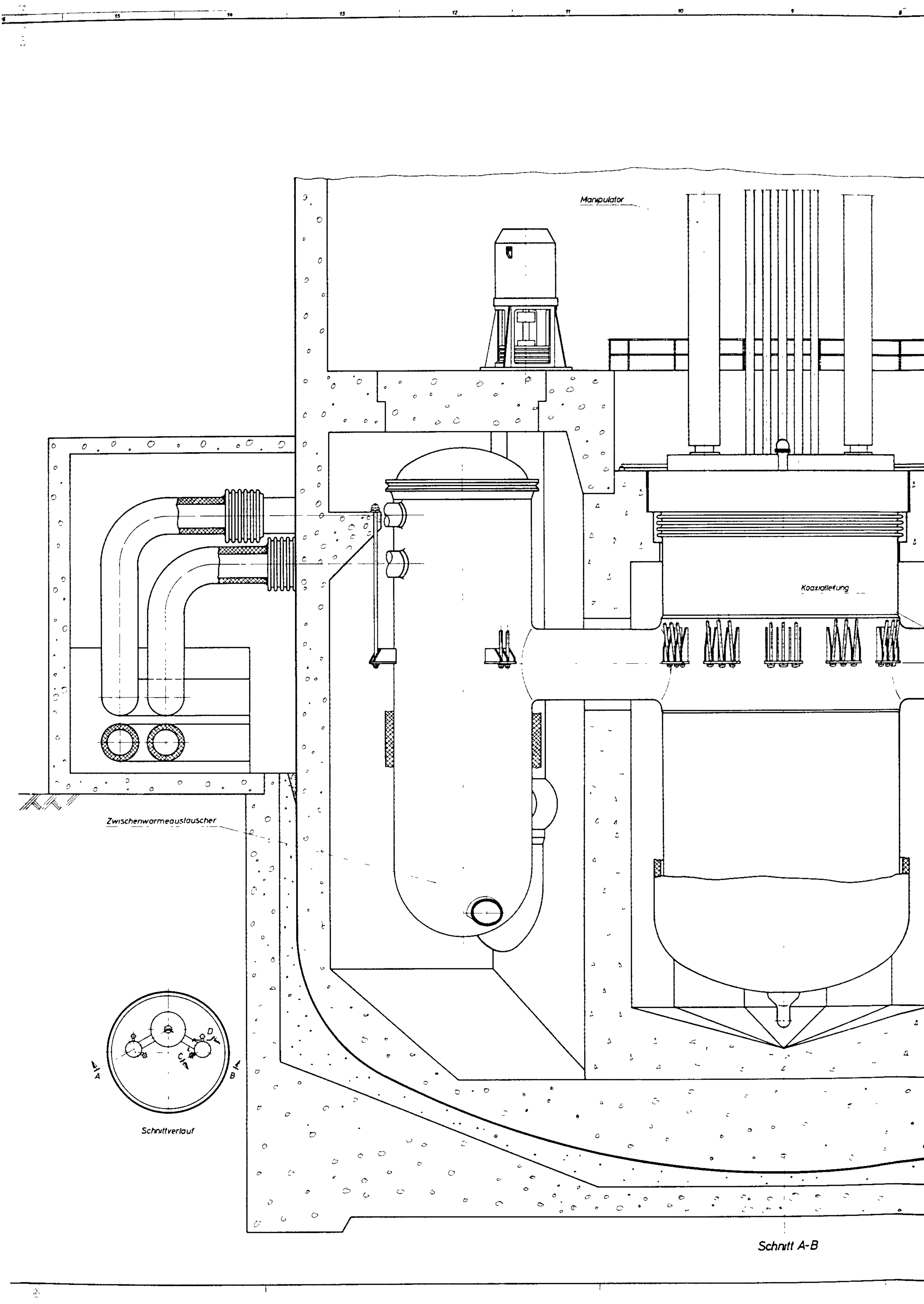
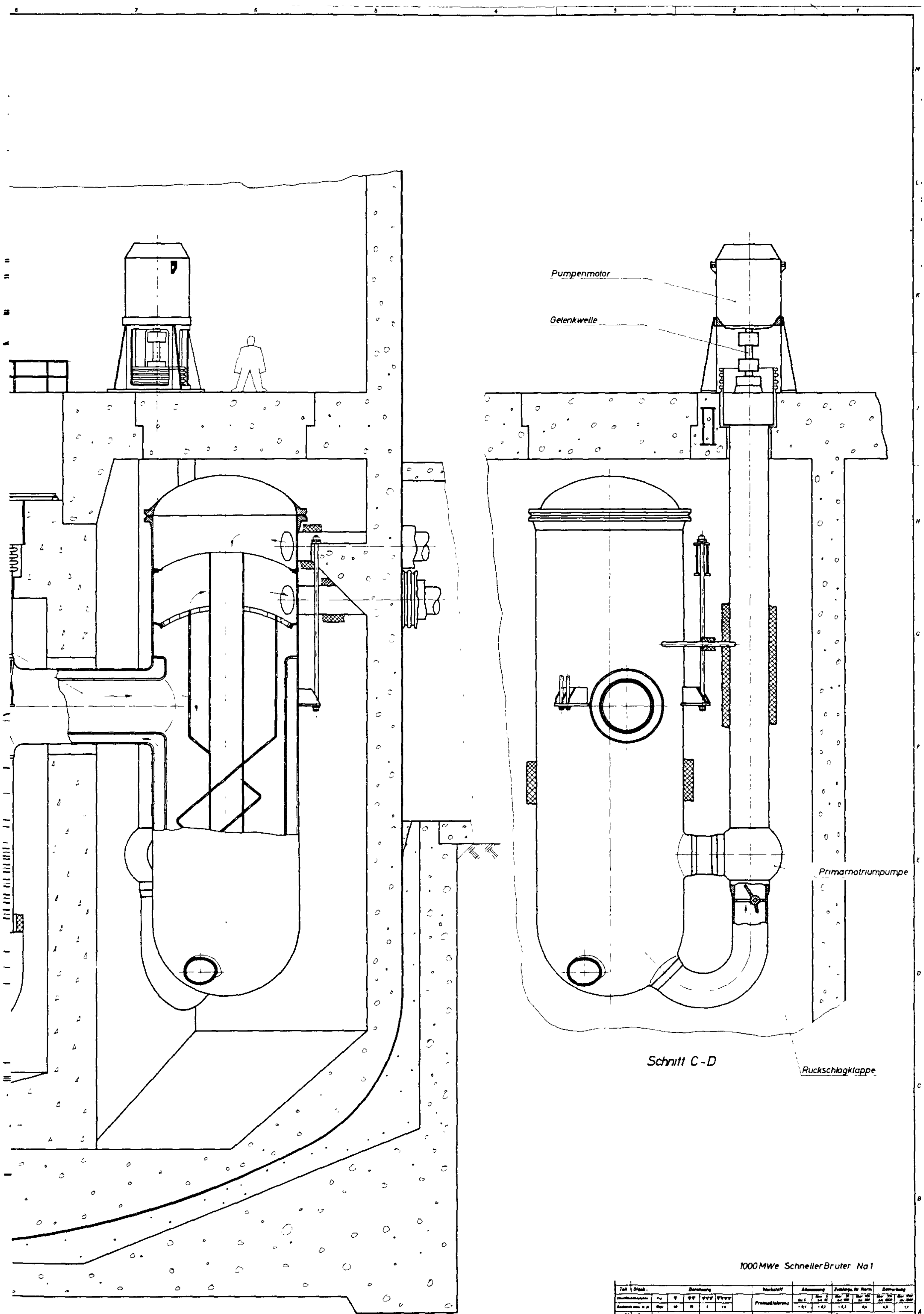


Abb. 2 c Brennstoffwechsel unter Natrium


56 000
28 000
17 500
Schutzbehälter
Manipulator
Steuerstabantriebe
BE-
Transport=
flasche
Pumpenmotor
Schutzbehälter=
belüftung
Materialschleuse


13 500
15 000
Personenschleuse
Kühlstoffleitung
Leckfühler
Reaktorkern
Traganker
Spornführung
Reaktorbehälter
Meßblock
Wärmeisolation
Kühlmittelauf = fangschale
0 1 2 3 4 5m
1000 MWe Schneller Brüter No 1
Gesellschaft für Kernforschung m. b. H.
7500 Karlsruhe
Postfach 947
1964
18. 12
1:50
Vertikalschnitt
durch das Reaktorgebäude
3


Manipulator
Koaxialleitung
Zwischenwarmeaustauscher
A
B
C
D
Schnittverlauf
Schnitt A-B



1000 MWe Schneller Bruter Na 1

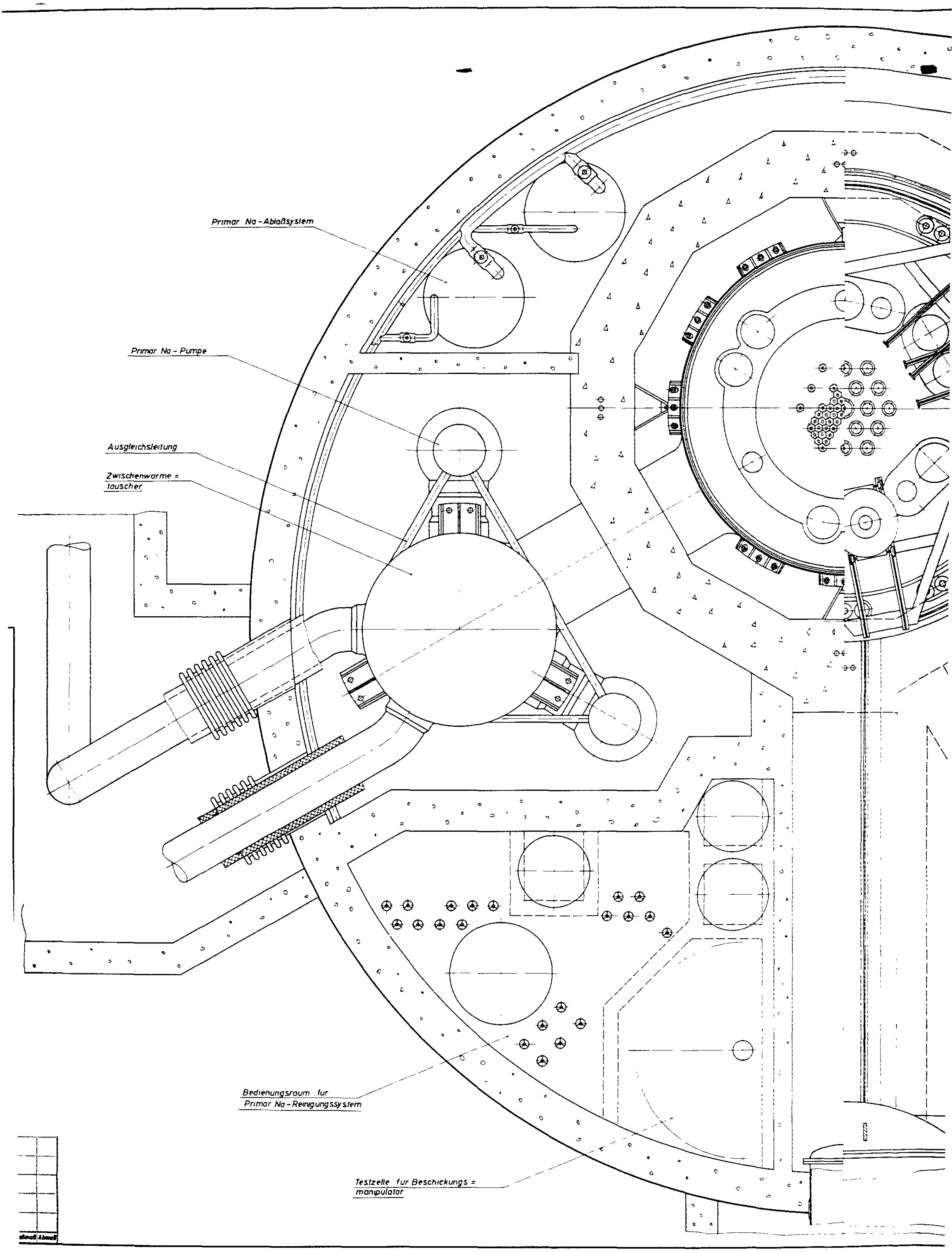
Gesellschaft für Kernforschung m. b. H
7500 Karlsruhe
Postfach 947

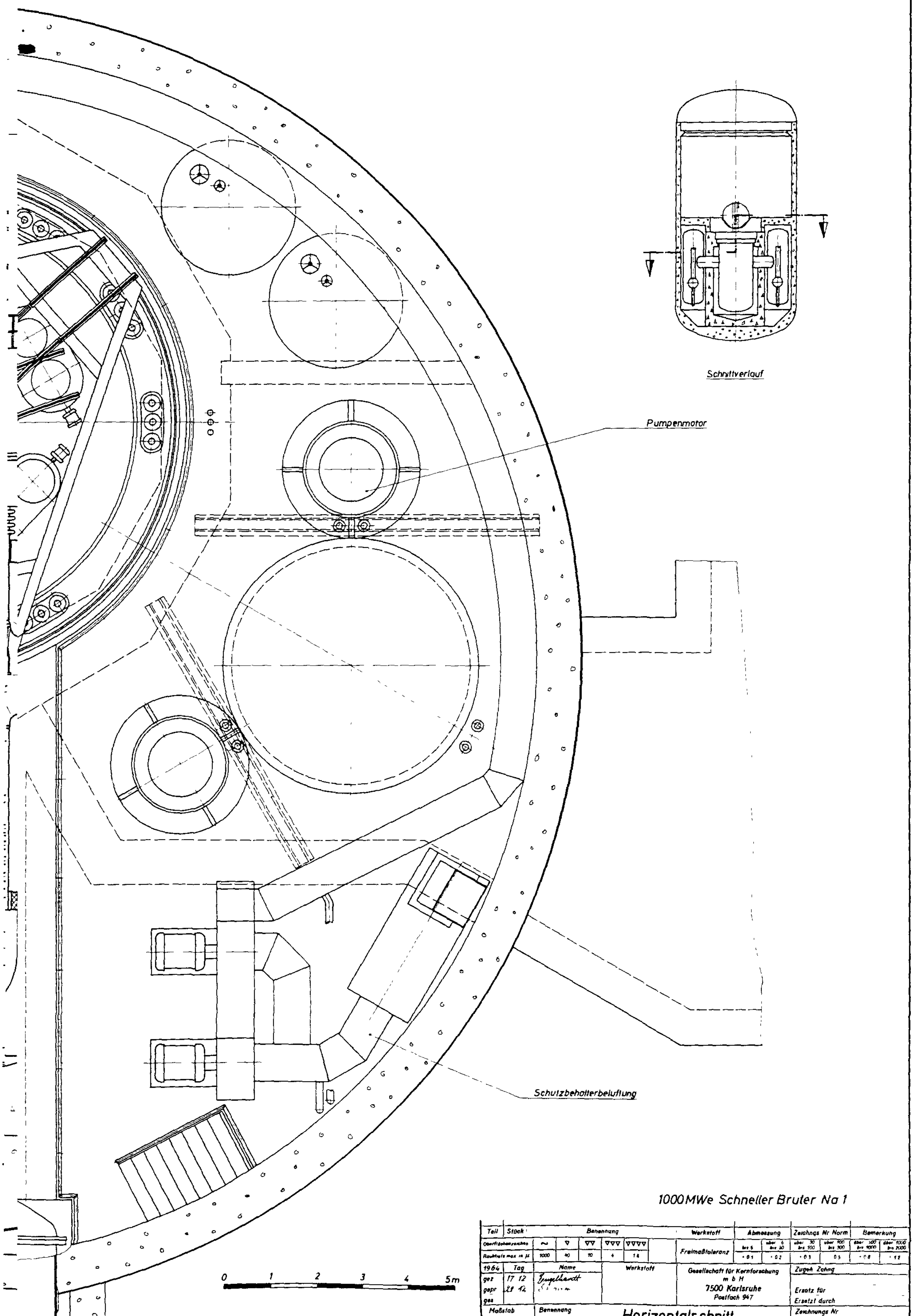
1964 gez 22.12

Maßstab 1:50

Benennung: Vertikalschnitt durch die Primärzelle

Zeichnungs Nr 4


Primar Na - Ablaßsystem
Primar Na - Pumpe
Ausgleichsleitung
Zwischenwarme =
tauscher
Bedienungsraum fur
Primar Na - Reinigungssystem
Testzelle fur Beschickungs =
manipulator
Abmaß



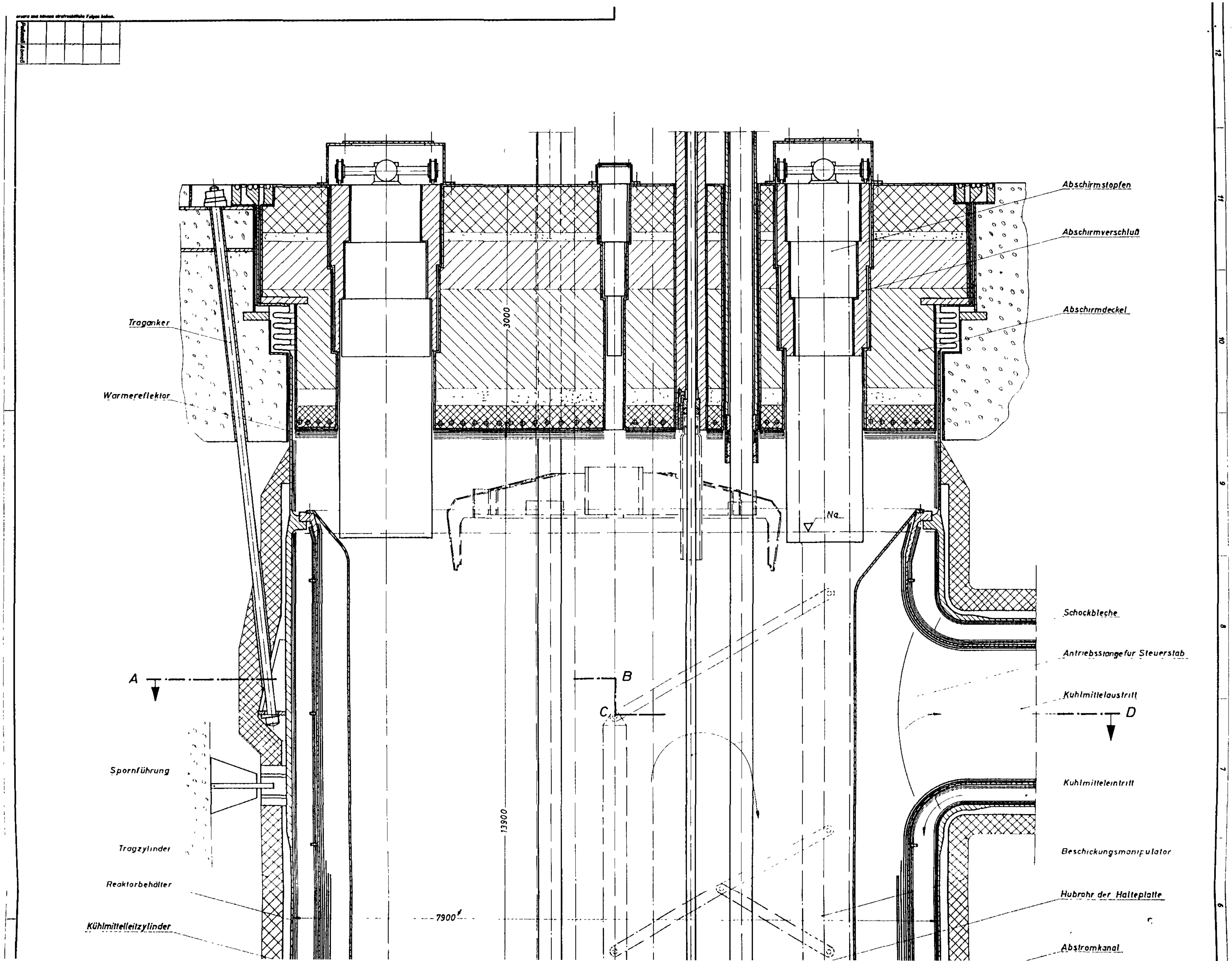
1000MWe Schneller Bruter Na 1

Teil	Stück	Benennung						Werkstoff	Abmessung		Zeichngs Nr Norm		Bemerkung	
Oberflächenzeichen		~	▽	▽▽	▽▽▽	▽▽▽▽		Freimaßtoleranz	bis 6	über 6 bis 30	über 30 bis 100	über 100 bis 300	über 300 bis 1000	über 1000 bis 2000
Rauhtiefe max in µ		1000	40	10	4	1,6			± 0,1	± 0,2	± 0,3	0,5	± 0,8	± 1,2

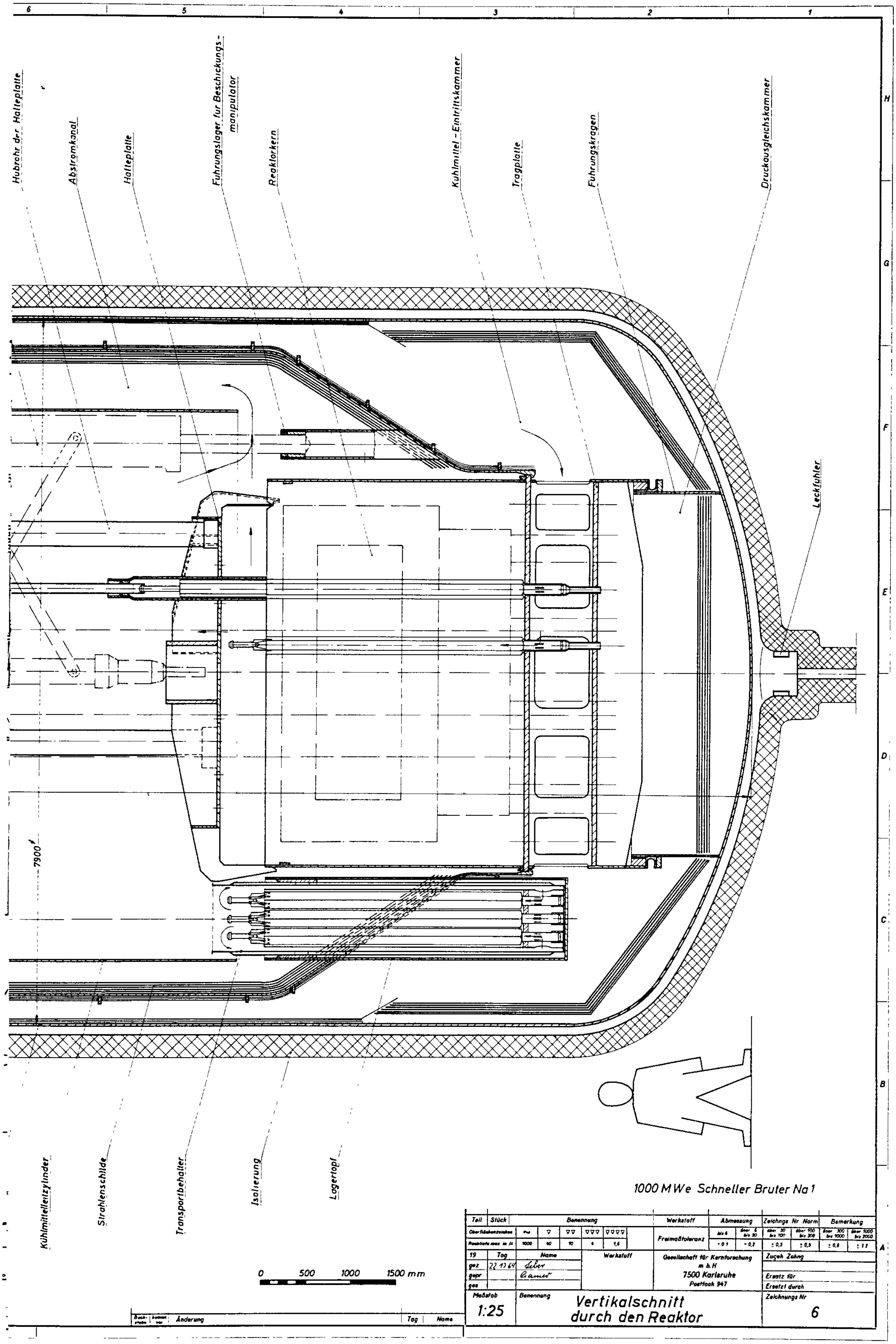
1964	Tag	Name	Werkstoff	Gesellschaft für Kernforschung m b H 7500 Karlsruhe Postfach 947	Zugeh. Zchng
gez	17 12	Engelhardt			
gepr	28 12				Ersatz für
ges					Ersetzt durch

Maßstab	Benennung	Zeichnungs Nr
1.50	Horizontalschnitt durch das Reaktorgebaude	5

Buch-stabe	komml. vor	Änderung	Tag	Name

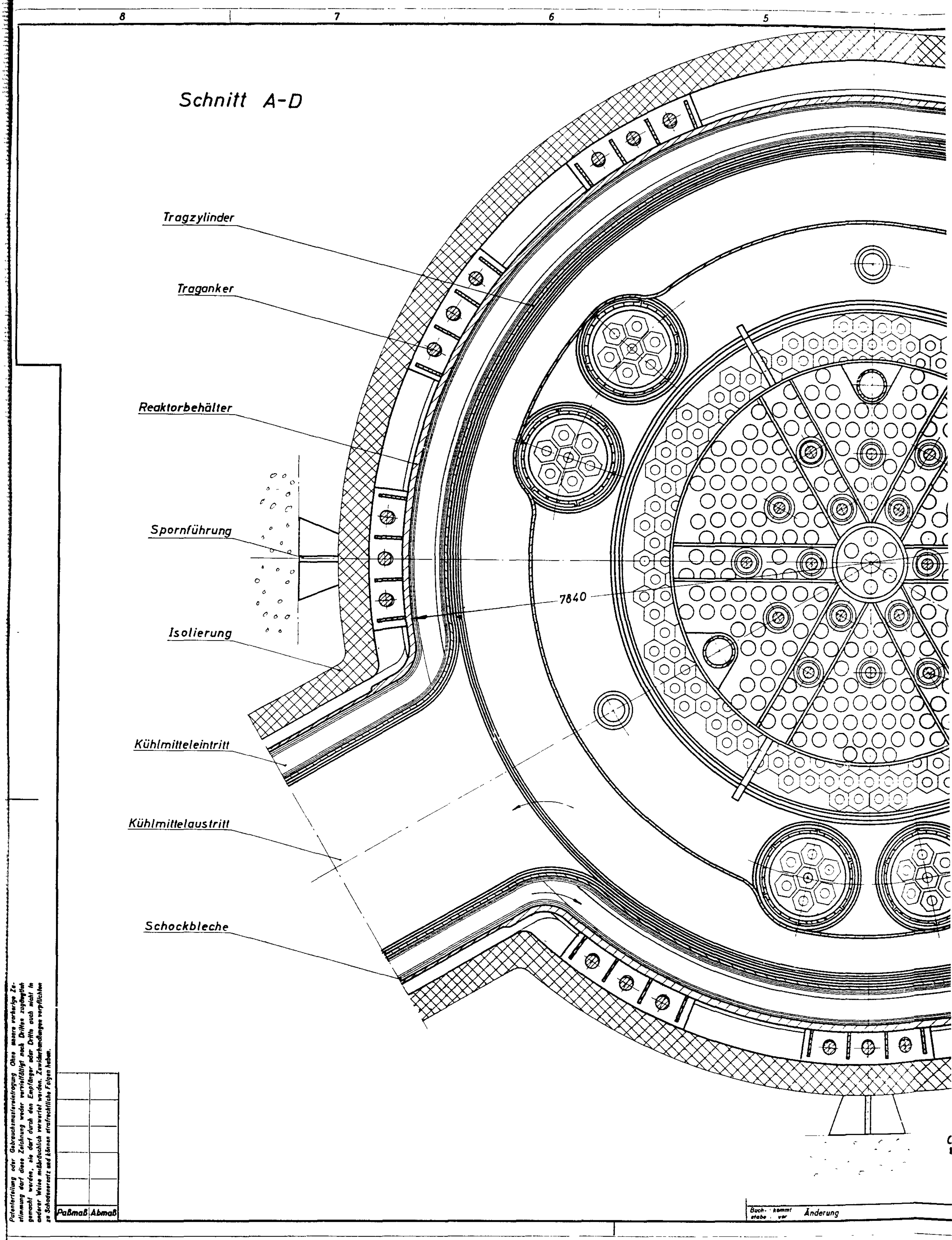


Paßmaß
Abmaß
12
11
10
9
8
7
6
Abschirmstopfen
Abschirmverschluß
Abschirmdeckel
Traganker
Warmereflektor
3000
Na
Schockbleche
Antriebsstangefur Steuerstab
Kuhlmittelaustritt
Kuhlmitteleintritt
Beschickungsmanipulator
Hubrohr der Halteplatte
Abstromkanal
A
B
C
D
13900
7900
Spornführung
Tragzylinder
Reaktorbehälter
Kühlmittelleitzylinder

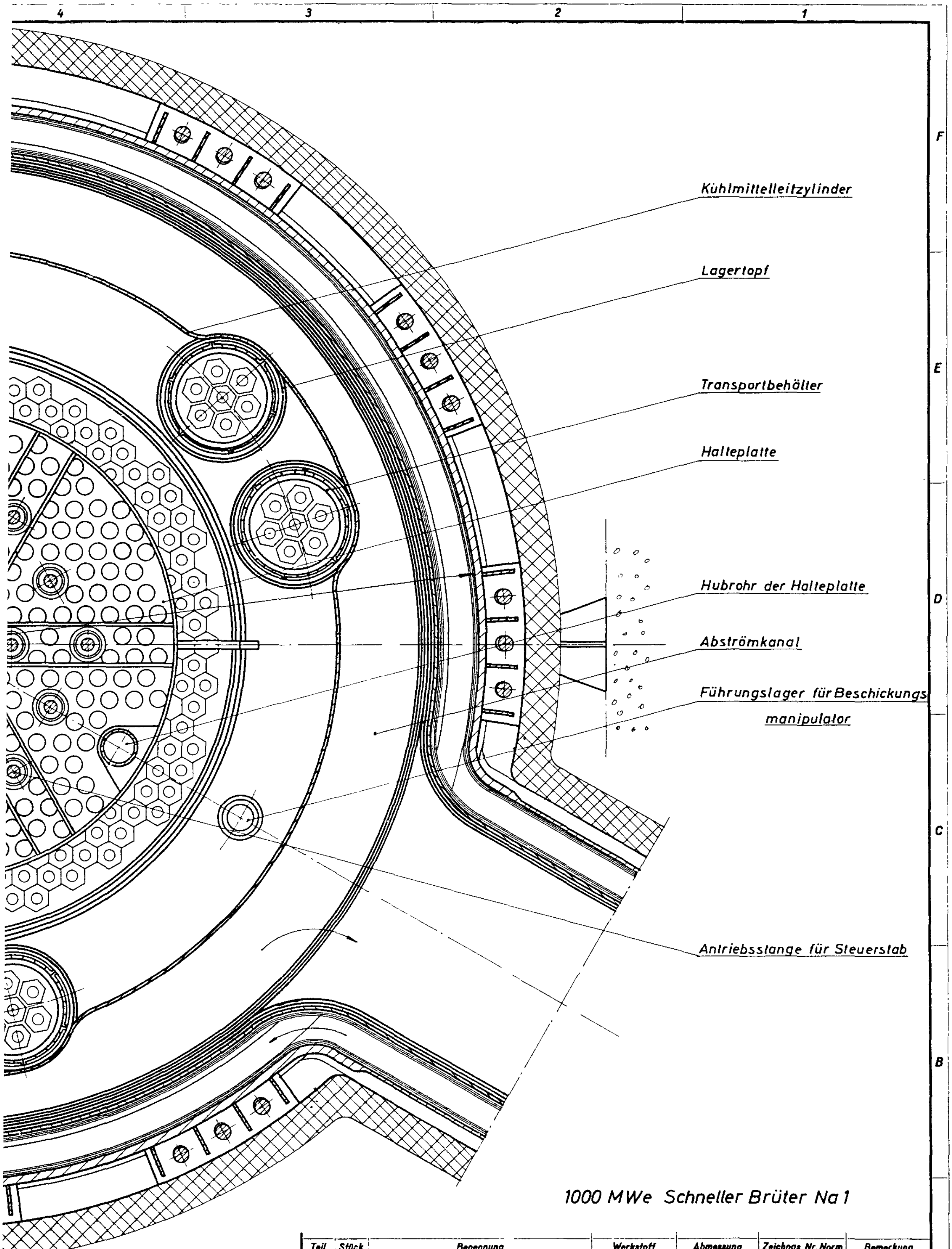


Hubrohr der Halteplatte
Abstromkanal
Halteplatte
Führungslager für Beschickungs-manipulator
Reaktorkern
Kühlmittel-Eintrittskammer
Tragplatte
Führungskragen
Druckausgleichskammer
Leckfühler
7900
Kühlmittelleitzylinder
Strahlenschilde
Transportbehälter
Isolierung
Lagertopf
0 500 1000 1500 mm
1000 MWe Schneller Bruter Na 1
Teil
Stück
Benennung
Werkstoff
Abmessung
Zeichngs Nr Norm
Bemerkung
Oberflächenzeichen
Rauhtiefe max in μ
1000
40
10
4
1,6
Freimaßtoleranz
bis 6
über 6 bis 30
über 30 bis 100
über 100 bis 300
über 300 bis 1000
über 1000 bis 2000
± 0,1
± 0,2
± 0,3
± 0,5
± 0,8
± 1,2
19
Tag
Name
gez
22 12 64
Leber
gepr
Camer
ges
Werkstoff
Gesellschaft für Kernforschung m.b.H
7500 Karlsruhe
Postfach 947
Zugeh. Zchng
Ersatz für
Ersetzt durch
Maßstab
1:25
Benennung
Vertikalschnitt durch den Reaktor
Zeichnungs Nr
6
Änderung
Tag
Name

Schnitt A-D



8
7
6
5
Tragzylinder
Traganker
Reaktorbehälter
Spornführung
Isolierung
Kühlmitteleintritt
Kühlmittelaustritt
Schockbleche
7840
Paßmaß
Abmaß
Buch-stabe
kommt vor
Änderung



1000 MWe Schneller Brüter Na 1

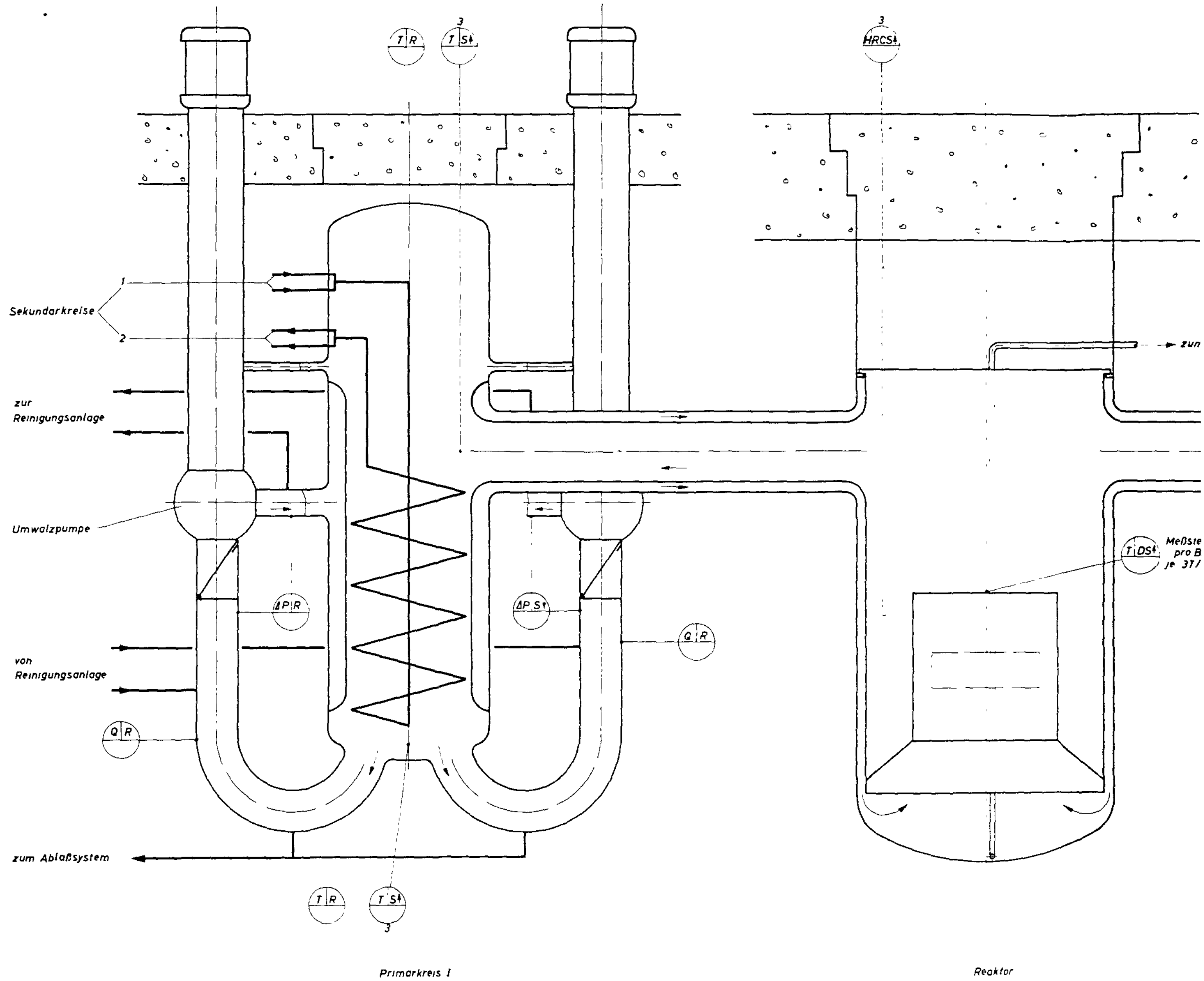
0 500 1000 1500 mm

Teil	Stück	Benennung					Werkstoff	Abmessung		Zeichngs Nr. Norm		Bemerkung	
Oberflächenzeichen		∾	▽	▽▽	▽▽▽	▽▽▽▽	Freimaßtoleranz	bis 6	über 6 bis 30	über 30 bis 100	über 100 bis 300	über 300 bis 1000	über 1000 bis 2000
Rauhtiefe max in µ		1000	40	10	4	1,6		±0,1	±0,2	±0,3	±0,5	±0,8	±1,2

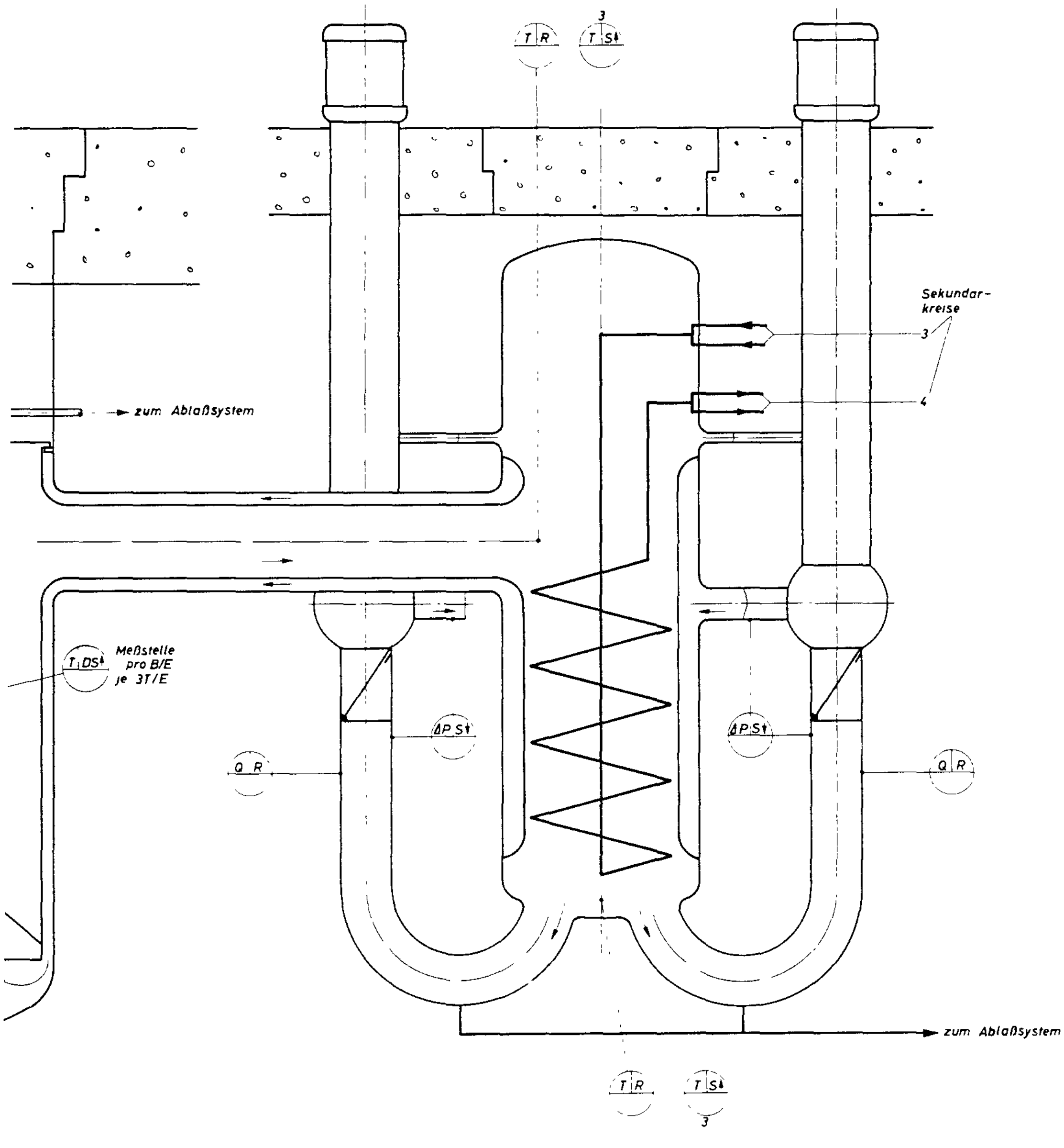
19	Tag	Name	Werkstoff	Kernreaktor Bau- u. Betriebs-Gesellschaft m. b. H. Karlsruhe	Zugeh. Zchng.
gez.	22.12.64	Leber			Ersatz für
gepr.		Cramer			Ersetzt durch
ges.					

Maßstab	Benennung	Zeichnungs Nr.
1:25	Horizontalschnitt durch den Reaktor	7

Tag Name



Sekundarkreise
1
2
zur
Reinigungsanlage
Umwalzpumpe
von
Reinigungsanlage
zum Ablaßsystem
T R
T S
HRCS
ΔP R
ΔP S
Q R
T DS
Primarkreis I
Reaktor



Primarkreis II

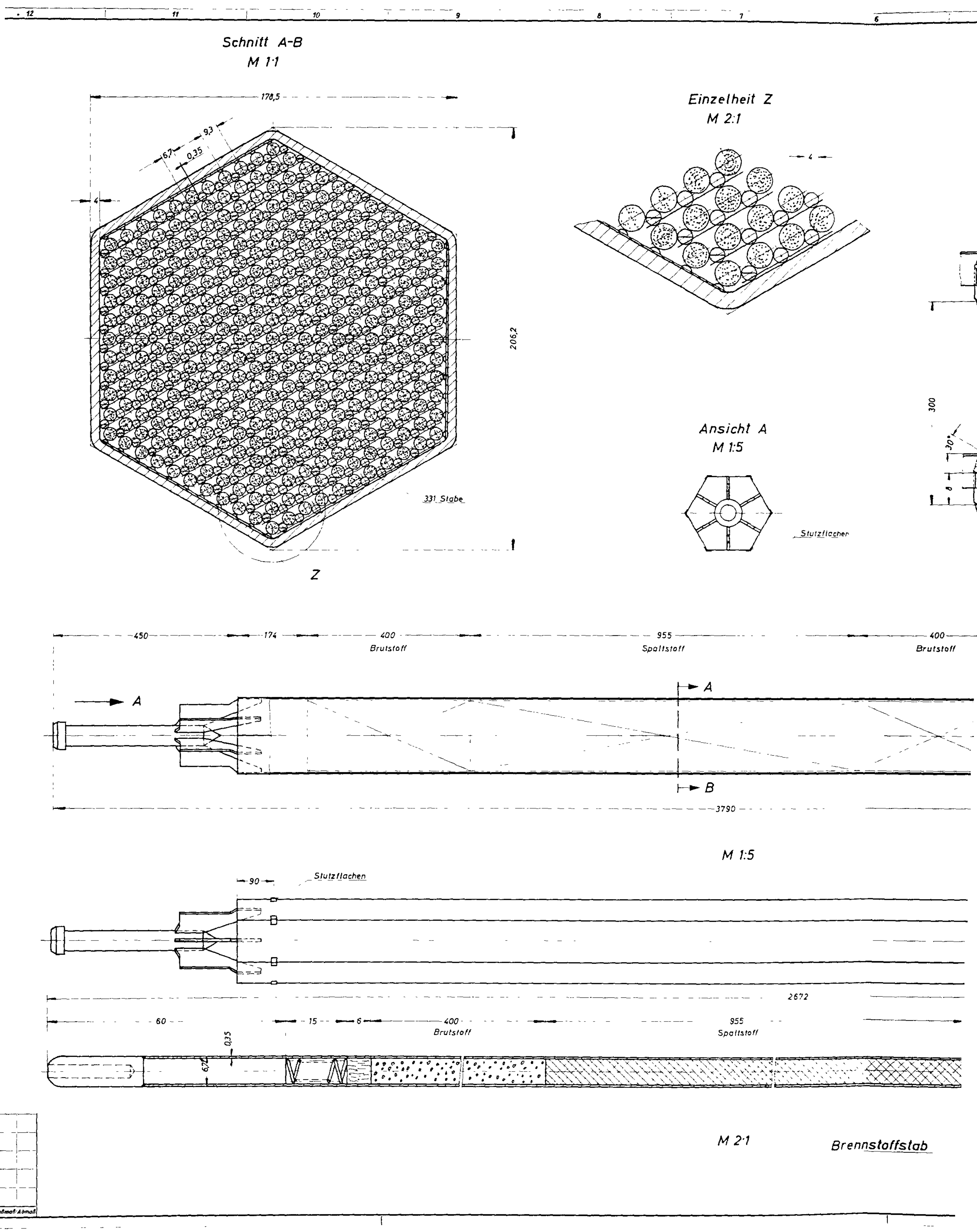
1000 MWe Schneller Bruter Na 1

Teil	Stück	Benennung						Werkstoff	Abmessung		Zeichngs Nr Norm		Bemerkung	
Oberflächenzeichen		~	∇	∇∇	∇∇∇	∇∇∇∇		Freimaßtoleranz	bis 6	über 6 bis 30	über 30 bis 100	über 100 bis 300	über 00 bis 1000	über 1000 bis 2000
Rauhtiefe max in μ		1000	40	10	4	1 6			± 0,1	± 0 2	± 0,3	± 0 5	± 0,8	± 1 2

1964	Tag	Name	Werkstoff	Gesellschaft für Kernforschung m b H 7500 Karlsruhe Postfach 947	Zugeh Zchng
gez	21 12	[signature]			Ersatz für
gepr					Ersetzt durch
ges					

Maßstab	Benennung	Zeichnungs Nr
	Schema des Primärsystems	8

Buch stabe	kommt vor	Änderung	Tag	Name



Schnitt A-B
M 1:1
178,5
9,3
6,7
0,35
4
206,2
331 Stabe
Z
Einzelheit Z
M 2:1
4
Ansicht A
M 1:5
Stutzflachen
300
30°
8
450
174
400
Brutstoff
955
Spaltstoff
400
Brutstoff
A
B
3790
M 1:5
90
Stutzflachen
2672
60
15
6
400
Brutstoff
955
Spaltstoff
0,35
6,7
M 2:1
Brennstoffstab
Paßmaß Abmaß