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by

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SAFEGUARDING FISSILE MATERIAL FLOW AT STRATEGIC POINTS IN POWER REACTORS

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1. Introduction

As opposed to the other branches of a fuel cycle industry, the fissile material is contained in well defined geometrical forms like fuel pins, in a power reactor of the presently known types. Under normal conditions, any change in the fissile material composition or concentration during its passage through the reactor, occurs inside the fuel pins and the fissile material cannot be lost directly in a waste stream as in the case of a reprocessing or a fabrication plant. In case it could be ensured that the pins or subassemblies containing the fissile material remain intact during their stay inside the reactor and that only those pins or subassemblies which have been identified at the entrance of a reactor can enter the reactor vessel for irradiation and leave it after a certain interval of time, no other safeguard measures would be required to control the fissile material flow through a reactor power station. This means only those measures which are required to ensure containment would be sufficient.

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1.1 It is however, possible, at least in principle, to establish a fissile material balance also in a reactor. The fissile material content of the subassemblies entering or leaving a reactor may be determined with varying degrees of accuracy. A detailed analysis of the possibility of establishing a material balance on the basis of reactor data indicate, that they are inaccurate to the range of ± 5 - 15 %. This point has been analysed in some detail in App. I. Measures based on material balance alone cannot be regarded as effective enough for safeguarding purposes, although some of the data used for this purpose might be utilized as independent information for implementing containment measures.

1.2 The present analysis indicates that power reactors of the presently known types are well suited for the safeguarding of fissile material mainly with the help of containment measures.

2. Common features of safeguards in power reactors of different types

Four reactors were chosen to evaluate the possibility of realizing various containment measures at strategic points. The general characteristics of these reactors are given in table I.

Table I: Reactors chosen for safeguards studies

Name	Type	Location	Supplier	Present state (1968)
CNA-Atucha	D_2O cooled and moderated, nat.U, continuous fuel loading	Argentina	Siemens AG. West Germany	Expected start-up 1971
Kernkraftwerk Obrigheim (KWO)	PWR, 3.0 % U, batch loading	W.Germany	Siemens AG. West Germany	Expected start up 1968 end
Kernkraftwerk Würgassen (KWW)	BWR, 2.1 % U, batch loading	W.Germany	AEG West Germany	Expected start up 1971
Kernkraftwerk Lingen (KWL)	BWR, 2.2 % U, batch loading	W.Germany	AEG West Germany	In full power operation since July 1968

The layout of the relevant reactor parts, the **movements** of the fuel elements and the possible safeguards measures for these power stations have been discussed in chapters 3 and 4 of this paper. Inspite of the apparent differences in the reactor designs chosen for this purpose, some common features are evident from the point of view of safeguarding fissile material. These features are summarized below. It should however, be noted that they are applicable to the reactor designs considered in this paper.

2.1 Fuel element movement in a nuclear power station takes place through three distinct containment areas, namely, the storage area for fresh fuel elements (dry storage), the reactor vessel, and storage pond for irradiated fuel elements (wet storage). Fresh fuel elements can always be stored in wet storage area, but irradiated fuel elements cannot be stored in dry storage area without heavy shielding.

2.2 Movement of fuel elements between two containment areas can be effectively monitored and registered both for on-load and off-load discharging reactor types.

2.3 With a combination of three safeguards measures, a) identification and counting of fuel elements in the two storage areas, b) registration of the movement and loading of the fuelling machines and of the reactor bay crane and c) registration of the activity level in the reactor bay, the movement and location of fuel elements between the three containment areas can be uniquely established as a function of time.

2.4 The three containment areas represent also the three strategic points at which all the safeguards measures are confined. However, the presence of inspectors, if at all, is required only at the storage areas and only during the transport periods. To safeguard fuel movement, presence of inspectors is not required in the reactor bay, or in the wet storage area during the refuelling procedure itself.

2.5 Records of integrated power output and some other general operation data from a reactor power station may be utilized as redundant information but are not essential for fulfilling safeguard duties.

2.6 From the point of view of safeguard, only limited information on the layout of a reactor power station is required. They are:

a) Location and overall layout of the three containment areas.

b) Transfer mechanisms for fuel elements from and to the reactor vessel and the two other containment areas, as well as, between the containment areas and outside world.

c) Number and function of all the machines and apparatus in the reactor bay which can be used to remove fuel elements from the reactor.

d) **Maximum possible number of fuel subassemblies for a core and the weight of each subassembly.**

3. Fuel movement in the four power stations

In this chapter, the fuel movements between the three containment areas have been discussed for the four nuclear power stations mentioned in table I. Technical data relevant from the point of view of safeguarding, have been summarized in table II for CNA-Atucha, in table III for KWO and in table IV for KWW and KWL. Besides the normal technical data on reactors and subassemblies,data on the two storage areas as well as the types and number of machines available for subassembly transfer between these containment areas have also been included in these tables. Fig. 1 shows the layout of the three containment areas and the fuel transfer mechanisms between these areas for the Atucha reactor. They are shown in Fig. 2 for the KWO, in Fig. 3 for KWL and in Fig. 4 for KWW. The broad outlines of the fuel subassemblies for Atucha and KWO are shown in Fig. 5. Fig. 6 gives a slightly more detailed drawing of the KWL subassembly. The subassembly for KWW has not been shown as it is very similar to that for KWL.

3.1 Fuel movement in Atucha reactor (Fig. 1)

The heavy water natural uranium power reactor called Atucha, designed by Siemens AG. West Germany, will be supplied to Argentina. The reactor station will have an installed electrical power of 340 MWe and is expected to go into operation in 1971. This reactor type appears to be particularly interesting for safeguard studies because of its on-load fuelling system.

In this reactor, the dry and the wet storage areas are outside the reactor containment building. The fresh subassemblies are stored in the dry storage area which is also in the spent fuel building. Normally the reactor is re-fuelled on-load. The fresh subassemblies are lowered into the wet storage pond using a light hoist, and are introduced into the reactor bay through the same transfer system through which the irradiated subassemblies are removed (see below). Alternatively, however, it may be desirable at times, to move new fuel into the reactor while it is shut down. The fresh subassemblies can be introduced into the reactor bay through the material lock and can be inserted in the reactor vessel with the help of the reactor-bay crane. The crane can reach the reactor vessel only when the reactor is shut down, as only then the concrete roof, separating the crane from the reactor vessel can be removed.

Irradiated fuel elements under normal conditions are removed on-load with the help of the fuelling machine. The machine is positioned over the closure of a subassembly to be unloaded and flooded internally with high pressure heavy water (120 Atm.). The closure is opened remotely with the help of a gripping tool, the fuel element is lifted out of the reactor vessel and the closure is closed again. The fuelling machine with the subassemblies goes to the intermediate decay tubes after depressurization, discharges the subassembly into one of these and goes back to the next subassembly closure. The drying machine which was over the outlet/inlet point during this operation, comes over to the decay tube and picks up the discharged subassembly. This is then dried in an air stream (to remove D_2O) inside the drying machine and the unit is moved over to the outlet/inlet point. The subassembly is lowered into the H_2O filled tilting flask after which the drying machine becomes free to take over the next subassembly. The tilting flask containing the subassembly is brought to a horizontal position and coupled to the fuel transfer tube in the transfer bay. In the transfer tube, the subassembly is loaded into a roller carriage which carries it through a water lock filled with light water, to the wet storage area. The subassembly is taken out here with the help of a tilting fixture, erected to a vertical position and is stored in racks provided for this purpose. The wet and the dry storage areas are spanned by a 60 t crane which can lift transport casks, used for transporting irradiated fuel elements from the reactor-station to a reprocessing plant.

In case the fuelling machine is out of order, a 30 t lead flask with the necessary hoisting mechanism, which is kept permanently in the reactor bay, can be used to lift fuel elements directly out of the reactor when it is shut down. This flask can be brought to the wet storage using the RB crane, the material lock, a special truck and the spent fuel building crane.

3.2 Fuel movement in KWO reactor (Fig. 2)

The 300 MWe pressurized water, slightly enriched uranium reactor at Obrigheim (KWO) has been designed by the firm Siemens AG. The **erection** of this reactor is virtually complete and the station should go into operation by the end of 1968. It is typical of the PWR type reactors which is expected to be **installed in the near future.**

The dry storage area of this power station is outside the reactor containment whereas, the wet storage area is inside it.

The fresh subassemblies are transfered from the dry storage, through the material air lock, to the wet storage area. They are lowered into the storage racks with the help of the fresh fuel lowering fixture. Normally, the fuelling machine is used to load the first core as well as for subsequent core loadings. But in principle, the reactor bay crane can also be used in conjunction with a simple fuel handling tool.

Irradiated subassemblies can be removed from the reactor vessel only under off-load condition. For this purpose, the reactor is shut down and after some cooling time, the **lid** of the reactor vessel is removed with the reactor bay (RB) crane. The space above the reactor is flooded with water and the level is brought up to that in the wet storage area. Before the refuelling begins, the upper core structure is lifted also with the RB crane and set aside. The sluice gate, which separates the reactor bay from the wet storage area is then removed. The fuelling machine containing a double gripper, lifts an irradiated subassembly from the reactor vessel, carries it under water to the wet storage and deposits it in one of the spent fuel storage racks. While returning, it picks up a fresh subassembly, carries it back to the reactor vessel and deposits it into the empty position caused by the removal of the subassembly. One of the grippers is used for the movement of subassemblies and the other, for the reshuffling of the control rods from one subassembly to the other. This reshuffling can be carried out either in the reactor vessel or in the wet storage area. Each time the reactor vessel is opened up for refuelling, about one third the total core loading is removed. In principle, the removal of irradiated subassemblies can also be done with the help of the RB crane. The **refuelling machine** is also used to load the spent fuel from the wet storage into a shielded cask. This cask is lifted by the RB-crane out of the pond and is transferred through the material lock to a reprocessing plant.

3.3 Fuel movement in KWW and KWL power stations (Fig.3 and 4)

Both the KWL and the KWW power stations are of the boiling water type and are designed by the firm AEG. The KWL station has an installed capacity of around 150 MWe from nuclear and about 90 MWe from fossile superheat and is located at Lingen, West Germany. The construction of this plant started in 1964 and it went into full power in July 1968. The 600 KWe KWW station will be erected at Würgassen, West Germany and is expected to go into operation in 1971.

Even though the basic principle of loading and unloading of fuel for these two boiling water reactors are the same, both of them have been chosen for the present analysis, as the layout of the three containment areas and the fuel routes inside the power station are somewhat different for the two reactors.

In the KWL station, the three containment areas are all located inside the reactor containment. The fresh subassemblies are brought inside the reactor containment through the material lock which is located at a height of 32 m from the ground level. They are picked up by the refuelling machine inside the containment and transfered to the dry storage. As in the case of KWO, the RB-crane can also be used for transferring the fresh subassemblies directly to the reactor bay for the first and subsequent core loadings, or for lifting the irradiated subassemblies from the core. But in normal practice, the fuelling and refuelling are always done with the fuelling machine.

The irradiated subassemblies can be discharged only after the reactor is shut down. The concrete roof of the reactor bay is first removed with the RB crane and then the reactor head. The reactor bay is flooded with water up to the level of the adjoining wet storage area. The upper core structure is lifted with the RB crane and set aside in position 1.703. The sluice gate between the wet storage and the reactor bay is removed and the fuelling machine starts transferring the irradiated subassemblies under water from the reactor core to the wet storage. On its way back, it picks up a fresh subassembly from the wet storage racks and inserts it into the core in the empty space caused by the removal of a subassembly. In this case also, about one **fourth** the total number of subassemblies is removed per discharge. No reshuffling on control rods from one subassembly to the other is carried out as the control rods are placed between the subassemblies and are not shifted radially during the life time of a core.

The fuelling machine is equipped with a light crane for transferring fresh subassemblies from the dry to the wet storage area and a gripper for lifting irradiated subassemblies from the reactor core and transporting them to the wet storage.

The method of filling the shielded cask with irradiated subassemblies, for transporting them from the reactor to a reprocessing plant, is the same as that for the KWO station.

The KWW power station uses a pressure suppression system for the reactor containment. For this reason, both the dry storage and the wet storage areas are outside the reactor containment. Subassembly movement mechanisms are however, very similar to those in the KWL station.

The fresh subassemblies are lifted to a height of 39.5 m (operation platform for the reactor bay) with the help of the RB crane, and transferred to the dry storage area. A light crane in the dry storage area is used to arrange these subassemblies in the racks. The fuelling machine carries the fresh subassemblies from the dry to the wet storage area.

As in the case of the other light water reactors discussed here, refuelling can be done only under off-load condition in this station also. For this purpose however, the top part of the reactor containment has to be removed with the RB-crane. After the removal of the reactor vessel head, a cylindrical steel structure is placed on top of the containment to facilitate flooding of the area above the containment and to raise the water level to that of the adjacent wet storage. The removal of the irradiated subassemblies with the fuelling machine starts after the sluice gate between the wet storage and the flooded area has been removed. Rest of the subassembly movements is similar to those used in the KWL station.

4. Safeguards measures

All the technical data, which appear to be relevant for safeguards studies, have been summarized in table V for the four power stations. It is to be noted that these reactors are all fuelled either with natural uranium or with slightly enriched uranium. Because of this, the fresh subassemblies themselves cannot be of great interest as objects of diversion. However, their importance increases immediately on irradiation in the reactor, because of the production of plutonium.

4.1 First safeguard measure

The first safeguard measure has therefore, to consist of registration and identification of all the fuel subassemblies which enter the power station. In a closed fuel cycle this measure is somewhat easier to perform than for an isolated power station, as the production output of the fabrication plant can then be checked with the input of subassemblies in the power station.

The identification markings of subassemblies have to be as tamper resistance as practicable. The importance of this fact cannot be over-emphasized. Individual subassemblies cannot be physically tracked and verified for safeguarding during their life times in a reactor and the possibility of partial or complete replacement of a subassembly during this period cannot be completely eliminated. Therefore, proper identification and sealing is the only way of ensuring that the same subassembly identified at the entrance of a reactor, also leaves the reactor after the irradiation and cooling period.

The first safeguard measure can best be carried out in the dry storage area where the fresh subassemblies are stored before their transfer to the wet storage or the reactor bay area. The dry storage area can therefore, be regarded as the first strategic point.

The same safeguard measure of registration and identification has to be performed at the wet storage area, in which both fresh and spent subassemblies are stored. This area can be regarded as the second strategic point. It should be noted that in all the four power stations, the wet storage has a considerably higher capacity than the dry storage.

In the wet storage area, an additional safeguard action has to be performed when the spent subassemblies are loaded into a cask for transporting them to a reprocessing plant, namely, the sealing of the cask. However, this measure will not be necessary in case the irradiated subassemblies are permanently stored in this area.

4.2 Other safeguard measures

Introduction: In case it could be assured or assumed that,

a) only those subassemblies which are registered and identified at the strategic point 1 can be irradiated in the reactor

and

b) all the irradiated subassemblies have to appear in the wet storage area,

then only the first safeguard measure would be sufficient for controlling the flow of fissile material in reactors of the types considered here.

However, at the present stage, and with existing reactor facilities, the probability of irradiation of subassemblies or other fuel units, which have not been identified and in which plutonium could be produced, cannot be completely eliminated. Therefore, some additional measures to monitor the movement of fuel to and from the reactor vessel have to be taken. It should be noted that all monitoring measures fall under the category of containment measures. Before these measures are discussed, some characteristics of the four reactor systems should be mentioned.

a) In all the four reactors, subassemblies can be removed or inserted only from the top of the reactor vessel.

b) A number of well defined and easily identifiable steps have to be taken in the reactor bay area for inserting or removing a fuel subassembly.

c) Only a limited number of special lifting machines are available in the reactor bay area which can be used to carry out these steps.

d) Irradiation of fuel subassemblies in these reactors is always associated with an increase in radioactivity in these subassemblies. If an irradiated subassembly is lifted out of a reactor, the environment at the top of the reactor vessel is bound to show an increase over the bakcground activity, unless the lifted subassembly is shielded.

4.2.1 Second safeguard measure: The movement of the fuelling machine and the load which it carries during this movement, can be registered easily with the help of a distance-cum-load measuring instrument which can be mounted at a suitable point on the fuelling machine. A typical printout from such an instrument, which has been used to register the movement of the fuelling machine at the KWO power station, is shown in Fig. 7a and the illustrated version of the same printout in Fig. 7b. It shows the distance-load diagram during the loading of some subassemblies in the reactor vessel. The lower part shows the depths travelled and the upper part, the loads carried by the fuelling machine for the corresponding depths. The abscissa gives the time. In this diagram the machine travelled back and forth between the reactor vessel and the wet storage.

If the reactor bay crane be also fitted with a similar instrument, the printout from this instrument would indicate whether the crane has been used for moving subassemblies from or to the reactor vessel. Thus, registration of all the lifting mechanisms in the reactor bay would provide a simple but effective means of monitoring the movement of fuels in the reactor bay, provided of course, such mechanisms have been used to move fuels.

Registration of the movements and loads carried by lifting mechanisms in the reactor bay and over the waste storage area, may therefore, be regarded as the second safeguard measure for these reactors.

4.2.2 Third safeguard measure: In case the regular lifting mechanisms are not used to take out an irradiated subassembly from the reactor, but some other provisional lifting structures are used for this purpose (which is theoretically possible), the distance-load instruments would not indicate the fuel movement. The third safeguard measure is proposed to monitor the fuel movement in such cases. This consists of a continuous registration of activity level in the reactor bay area on top of the reactor vessel. The sensing units have to be mounted and arranged in such a way that falsification of input data becomes difficult.

Any movement of the lifting mechanisms in the reactor bay over the reactor head - if associated with a subassembly movement - could mean only two things:

a) The subassemblies have been reshuffled inside the reactor core, or taken out for observation and put back into the same position; in this case all the moved fuel would have to go back to the reactor vessel and the number of subassemblies in the dry and the wet storage area would remain unchanged.

b) A loading or an unloading of subassemblies for the reactor core has taken place. In that case, each of the outgoing movement from the reactor vessel to the wet storage has to be associated with an increase in the number of subassemblies in the wet storage area. Similarly, the ingoing movements can be associated with a change in the number of subassemblies either in the dry or in the wet storage area. In a like manner, the increase in activity over the reactor vessel area may be checked with the change in the number of subassemblies in the dry or the wet storage.

Thus, a combination of these two measures can monitor the fuel movement fairly effectively between the three containment areas.

It should be noted that a part of the second safeguard measure and the third safeguard measure are carried out in the reactor bay area. Therefore, theoretically this area can be regarded as the third strategic point. However, it is important to realize, that the presence of inspectors are not required at this point to implement these measures.

5. Development work necessary

Of the three safeguards measures suggested in chapter 4, the second one, based on the registration of movement and loading history of all the relevant lifting mechanisms, is relatively simple to carry out. Standard instruments for this purpose are available in the market. However, some development work, to improve the sensitivity of the instrument and to make them tamper resistant, would be necessary.

A larger research and development effort would be required before the other two measures can be effectively implemented. The problems involved and the type of work necessary are discussed below.

5.1 Sealing and identification of subassemblies

It was indicated in 4.1 that sealing and identification of subassemblies is the most important of the three safeguards measures discussed. It may be noted that this measure is required at three nuclear facilities in the fuel cycle. The fresh subassemblies have to be sealed in a fabrication plant. These seals have to be checked and identified in the dry and wet storage areas of a nuclear power station, and finally, they have to be checked and identified once more in the wet storage of a reprocessing plant, in case the subassemblies are reprocessed.

Two types of development work are required. Firstly, development of the sealing of subassemblies, and secondly, development of the identification system in the wet storage area.

The main purpose of sealing is to ensure that the subassemblies remain intact during their passage between two strategic points. Without sealing, fuel pins can be introduced or replaced, with varying degrees of efforts, in the subassemblies (Figs.5, 6) for the four reactor types considered here. Sealing of these subassemblies have to be done in such a way that any tampering can be detected with a high degree of probability. It should remain intact and identifiable during the irradiation of subassemblies in the reactor and later on, during their storage in the wet storage area. It should be possible to reseal a subassembly in the wet storage area, under the filling medium (which is water in most cases) in case some damaged pins in an irradiated subassembly have to be replaced. Finally, it should be economic. For this purpose, development work has to be done in three areas, namely, design of different methods of sealing, fabrication of prototype sealed subassemblies and irradiation experiments with sealed subassemblies in a reactor.

The main problem of identification lies in remotely identifying the seals on irradiated subassemblies in the wet storage under the filling medium. A photographic under water identification system for irradiated subassemblies has been reported by Kindermann [1]. Further development on similar lines might be necessary for effective identification.

5.2 Measurement of activity in reactor bay

The detection and registration of activities of irradiated fuel subassemblies, in the flooded bay over the reactor vessel by fixed and sealed detectors, requires solution of a number of problems. The detectors must be sensitive enough to give a clear signal which should be significantly different to the actual background signal. They must be able to distinguish between a simulated and an actual signal and must be specific for irradiated fuel subassemblies. They should operate unattented over long periods of time.

Some idea on the required sensitivity of the detector can be obtained from Fig. 8. The activity of a fuel subassembly, which has been irradiated in a boiling water reactor at a constant thermal power of 1 MW, as measured under water by a detector placed at various distances, has been shown with decay time after shut-down as the parameter. For other subassembly powers, the values of Fig. 8 have simply to be multiplied with the actual thermal power in MW. As an example, the KWL subassembly with about 2 MW_{th} rating, would show an activity of around 4 mr/hr at a distance of 2.5 m, 9 days after shut down of the reactor. The background level of the water in the flooded area of the reactor bay, is around 10 mr/hr. The minimum possible distance between subassemblies and the detector range from 3 to 4 meters. This shows that for light water reactors considered here, important refinements over the presently known detection methods would be required. For the heavy water reactor, because of the absence of water in the reactor bay, the conditions which the detectors have to meet, would be less severe.

Gamma energy detectors appear to be the most suitable, although for under water detection, Cerenkov radiation might also be useful.

Any detector system, which might be developed, have to be tested over long periods of time under actual operating conditions in a reactor.

Literature

[1] Kinderman, E.M.: Development of Photographic Identification Systems for Fuel Elements. Stanford Research Institute, California. IAEA Research Contract No. 244 (1965)

Table II: Characteristics of the ATUCHA power station (preliminary)

1.	**General**	
1.1	Type	D_2O cooled and moderated, natural uranium, **pressurized**
1.2	Location	Argentina
1.3	Thermal power [MW_{th}]	1100
1.4	Subassembly loading mechanism	continuous under load 2 **subassemblies/d** under normal conditions
2.	**Subassembly**	
2.1	**Max. Number of subassemblies** in reactor core	253
2.2	Largest diameter of subassembly [mm]	107.8
2.3	Length of subassembly (in cooling medium) [mm]	5550
2.4	Total length of subassembly (including shielding length) [mm]	9500
2.5	UO_2-weight per subassembly [kg]	173
2.6	Total weight per subassembly [kg]	217
2.7	Number of fuel **pins per subassembly**	36
2.8	Fuel pin diameter [mm]	11.9
2.9	Total fuel pin length [mm] **(active length)**	5420

Table II [ctd.]

3. Reactor vessel

One cylindrical vessel welded to a round bottom, having 4 openings for cooling medium, and 3 for moderator.

One flanged top with 253 openings for cooling medium, 29 for control rods and a number of small holes for instrumentation.

253 fuel element closures with connecting rods.

One vessel support ring.

One complete insulation for the vessel surface.

Major dimensions

3.01	Internal diameter of the cylinder [mm]	5360
3.02	Largest outside diameter of the cylinder [mm]	7200
3.03	Cylinder height (without cover) [mm]	9860
3.04	External diameter of the cover [mm]	6200
3.05	Height of the cover [mm]	2260
3.06	Maximum height of the reactor pressure vessel (from the bottom to the top of the coolant opening) [mm]	12120
3.07	Weight of the lower part [t]	~ 320
3.08	Weight of the top cover [t]	~115
3.09	Nuts and bolts and spacers [t]	~ 25
3.1	Fuel element joints closures [t]	~ 20

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Table II /‾ctd._/

4.	Wet storage	
4.1	Storage dimensions L x W x H /‾m_/ 2 pits	10 x 5.2 x 16.5
4.2	Storage capability Number of subassemblies	1300
4.3	Filling medium	H_2O
5.	Dry storage	
	L x W x H /‾m_/	10 x 2.2 x 12
5.2	Storage capability Number of subassemblies	200
6.	Subassembly transfer machines	
6.1	Heavy crane for transport of fresh subassemblies from outside into the dry storage area	1(**with two hoisting mechanisms)** (60 + 5 t)
6.2	Heavy crane for removing shielding, refuelling machine, pressure vessel lid and irradiated subassemblies in a shielded cask from the reactor (for emergency purposes only)	1 (200 + 80 t)
6.3	Small crane for light duty inside reactor room	1 (1 t)
6.4	Light crane for handling fresh fuel elements within the dry storage	1 (1.4 t)
6.5	Fuelling machine, containing fuel element hoist for fuel transport between reactor vessel and wet storage (weight 60 t)	1 (1.2 t)
6.5.1	Light crane for bringing fresh fuel from dry storage to wet storage	1 (1 t)

Table III: Characteristics of the KWO power station

1.	**General**	
1.1	Type	H_2O cooled and moderated pressurized water reactor, enriched uranium
1.2	Location	Obrigheim, Germany
1.3	Thermal power [MW_{th}]	907.5
1.4	Subassembly loading mechanism	Batch loading once a year in flooded reactor pit
2.	**Subassembly**	
2.1	Number of subassemblies in reactor core	121
2.2	Cross section of subassembly [mm]	200 x 200
2.3	Length of active fuel [mm]	2750
2.4	Total length of subassembly [mm]	3175
2.5	UO_2-weight per subassembly [kg]	330
2.6	Total weight per subassembly [kg]	445
2.7	Number of fuel pins per subassembly	180
2.8	Fuel pin diameter [mm]	10.7
2.9	Total fuel pin length [mm]	2917

3. Reactor vessel

One cylindrical vessel welded to a round bottom, having 4 openings for cooling medium.

One flanged lid with 32 openings for control rod drive mechanisms, 16 for instrumentation, and 1 for venting.

One vessel support ring. One complete insulation for the vessel surface.

Table III [ctd.]

Major dimensions

3.1	Internal diameter of the cylinder [mm]	3270
3.2	Largest outside diameter of the cylinder [mm]	3800
3.3	Vessel height (without top) [mm]	7930
3.4	External diameter of the lid [mm]	3800
3.5	Height of the cover [mm]	1895
3.6	Maximum height of the reactor pressure vessel (from the bottom to the top of the control rod flanges) [mm]	9825
3.7	Weight of the lower part [t]	145
3.8	Weight of the vessel lid [t]	50
3.9	Weight of the vessel lid complete with control rod drive mechanisms and lifting fixture	80
3.10	Weight of the upper core structure complete with lifting fixture	25

4. Wet storage

4.1	Storage dimensions L x W x H [m]	8 x 6.2 x 11
4.2	Storage capability Number of subassemblies	162
4.3	Filling medium	H_2O and boric acid

5. Dry storage

	L x W x H [m]	10.7 x 4.8 x 4.8
5.2	Storage capability Number of subassemblies	48

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Table III /̄ ctd._7

6. Subassembly transfer machines

6.1	Turbine hall crane for loading of fresh or irradiated subassemblies from or onto outside truck and internal vehicles for storage areas	1 (80 + 15 t)
6.2	Heavy crane for removing shielding pressure vessel lid and irradiated subassemblies in a shielded cask from the wet storage	1 (200 + 32 t)
6.3	Light crane for handling fresh fuel elements within the dry storage	1 (2 t)
6.4	Fuelling machine, containing fuel element handling tool for fuel transport between reactor vessel and wet storage	1 (2 t)
6.5	Light crane for lowering fresh fuel into wet storage racks	1 (5 t)

Table IV: Characteristics of the KWL and KWW power stations

		KWL	KWW
1.	**General**		
1.1	Type	H_2O cooled and moderated, slightly enriched uranium	H_2O cooled and moderated, slightly enriched uranium
1.2	Location	Lingen, Germany	Würgassen, Germany
1.3	Thermal power [MW_{th}]	520	1912
1.4	Subassembly loading mechanism	Batch loading under off load condition	Batch loading under off load condition
2.	**Subassembly**		
2.1	Number of subassemlies in reactor core	284	444
2.2	Width of subassembly [mm]	114.1	134
2.3	Active length of subassembly [mm]	2935	3660
2.4	Total length of subassembly [mm]	3500	4470
2.5	UO_2-weight per subassembly [kg]	128	220
2.6	Total weight per subassembly [kg]	187	320
2.7	Number of fuel pin per subassembly	36	49
2.8	Fuel pin diameter [mm]	14.3	14.3

3. Reactor vessel

One cylindrical vessel welded to a round bottom, having 10 openings for cooling medium and 69 openings for bottom entry control rods.

One flanged top with openings for instrumentation.

One vessel support ring.

Table IV /‾ctd._/

		KWL	KWW
Major dimensions			
3.01	Internal diameter of the cylinder /_mm_/	3600	5450
3.02	Largest outside diameter of the cylinder /_mm_/	4500	6400
3.03	Cylinder height (without top) /_mm_/	9900	-
3.04	Maximum height of the reactor pressure vessel	14750	20500
3.05	Weight of the lower part /_t_/	175	372
3.06	Weight of the top cover /_t_/	35	78
3.07	Weight of internal core cover /_t_/	4	8
4.	Wet storage		
4.1	Storage dimensions		
	L x W x H /‾m_/	8 x 7 x 10	11.8 x 10 x 13.2
4.2	Storage capability		
	Number of subassemblies	426	640
4.3	Filling medium	light water	light water
5.	Dry storage		
	L x W x H /‾m_/	8 x 6.5 x 6.5	12 x 8.25 x 7.42
5.2	Storage capability		
	Number of subassemblies	88	136

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Table IV [ctd.]

		KWL	KWW
6.	Subassembly transfer machines		
6.1	Heavy crane for transport of fresh subassemblies from outside into the dry storage area	1 (10 t)	1 (15 t)
6.2	Heavy crane for removing shielding, containment cover, pressure vessel top and irradiated subassemblies in a shielded cask from the wet storage	1 (75 t)	1 (110 t)
6.3	Small crane for light duty	-	1 (0.6 t)
6.4	Light crane for handling fresh fuel elements within the dry storage	1 (1 t)	1 (1.8 t)
6.5	Fuelling machine, containing fuel element manipulator for fuel transport between reactor vessel and wet storage	1 (0.75 t)	1 (1.4 t)
6.5.1	Light crane for bringing fresh fuel from dry storage to wet storage	1 (0.5 t)	1 (0.8 t)

Table V: Relevant technical data on four power stations which influence safeguards measures

		Atucha	KWO	KWL	KWW
1.	Type	D_2O moderated and cooled, natural uranium	H_2O moderated and cooled, pressurized water, slightly enriched U	H_2O moderated and cooled, boiling water, slightly enriched U	H_2O moderated and cooled, boiling water, slightly enriched U
2.	Subassembly	Cylindrical; 36 fuel pins; no space for control rods; 1 empty tube for supporting spacers etc; disassembly only by sawing	Square; 180 fuel pins; 16 empty positions for control rods; disassembly only by sawing	Square; 36 fuel pins; no space for control rods; disassembly by unscrewing 8 bolts	Square; 49 fuel pins; no space for control rods; disassembly by unscrewing 12 bolts
3.	Dry storage area	Outside reactor containment; served by a 60 t crane; easily accessible from outside	Outside reactor containment; served by light crane; easily accessible from outside	Inside reactor containment; served by a 1 t crane; accessible only through material or personnel lock	Outside reactor containment; served by a 2 t crane; easily accessible from outside
4.	Wet storage area	Outside reactor containment; served by a 60 t crane; connected to reactor vessel through tilting flask and fuel transfer bay; accessible from outside	Inside reactor containment; served by fuelling machine, RB crane and fresh fuel lowering fixture; fuel movement from and to reactor vessel only by fuelling machine and RB-crane; accessible only through material and personnel lock	Inside reactor containment; served by fuelling machine and RB crane; fuel movement from and to reactor vessel with these two mechanisms; accessible only through material and personnel lock	Outside reactor containment; served by fuelling machine and RB crane; fuel movement from and to reactor vessel with these two mechanisms; easily accessible from outside

./.. continued

Table V(continued)

5. Reactor vessel -reactor bay	Accessible only through material or personnel air lock; served by fuelling machine; and RB crane after removal of concrete roof; outside reactor vessel intermediate storage possibility for subassembly	Accessible only through material or personnel lock; served by fuelling machine and RB crane	Accessible only through material or personnel lock; served by fuelling machine; and RB crane after removal of concrete roof	Accessible only through material or personnel lock; served by fuelling machine; and RB crane after removal of containment top
6. Refuelling possibility	On load and off load	Off load	Off load	Off load
7. Possibility of subassembly removal from				
a) Dry storage	From all sides	Only from the top	Only from the top	Only from the top
b) Wet storage	From top and from one side	From top and from one side	From top and from one side	From top and from one side
c) Reactor vessel	Only from the top	Only from the top	Only from the top	Only from the top

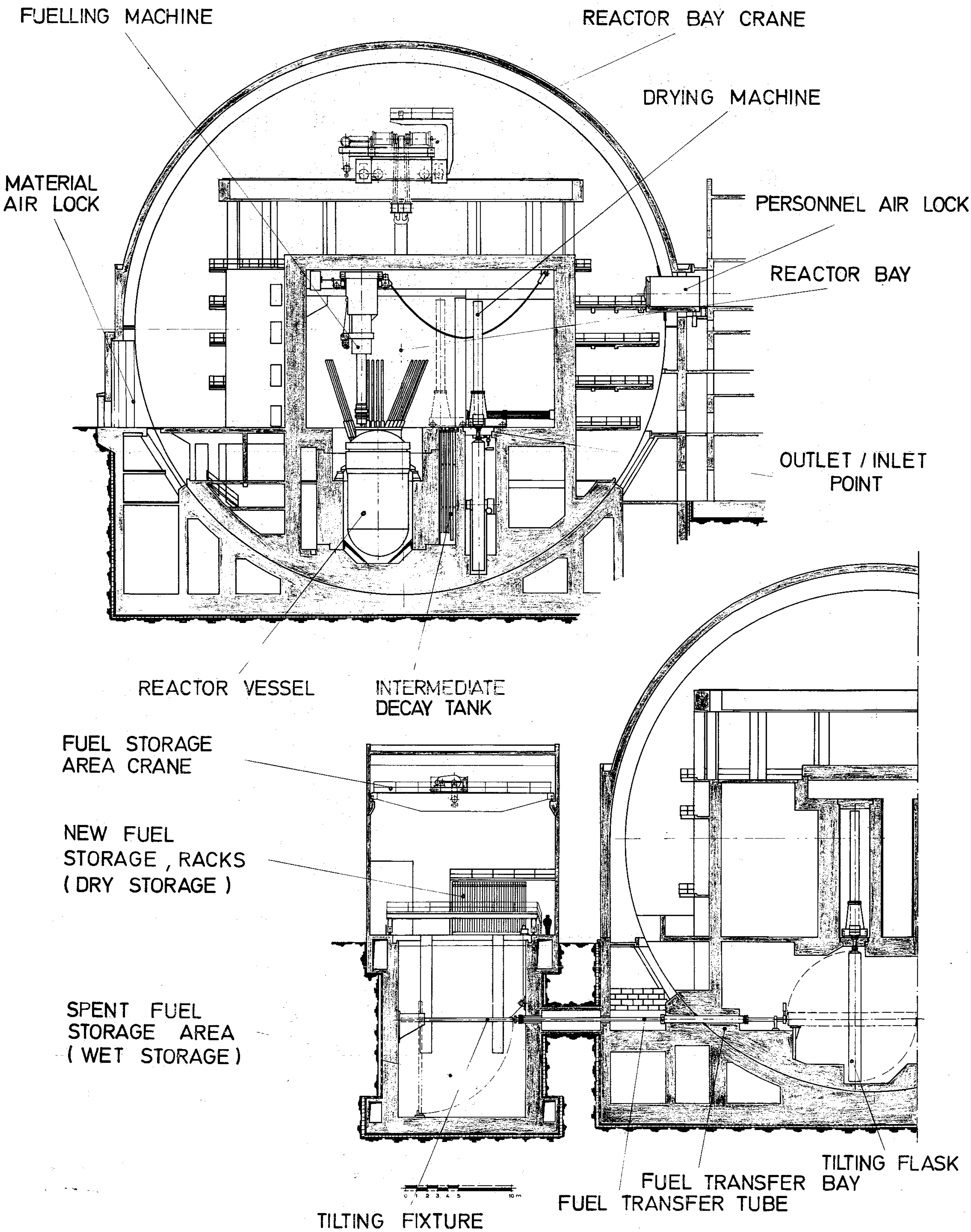


FIG. 1 GENERAL LAY-OUT OF DRY AND WET STORAGE AREAS, REACTOR BAY AND FUEL TRANSFER MECHANISMS FOR ATUCHA POWER STATION.

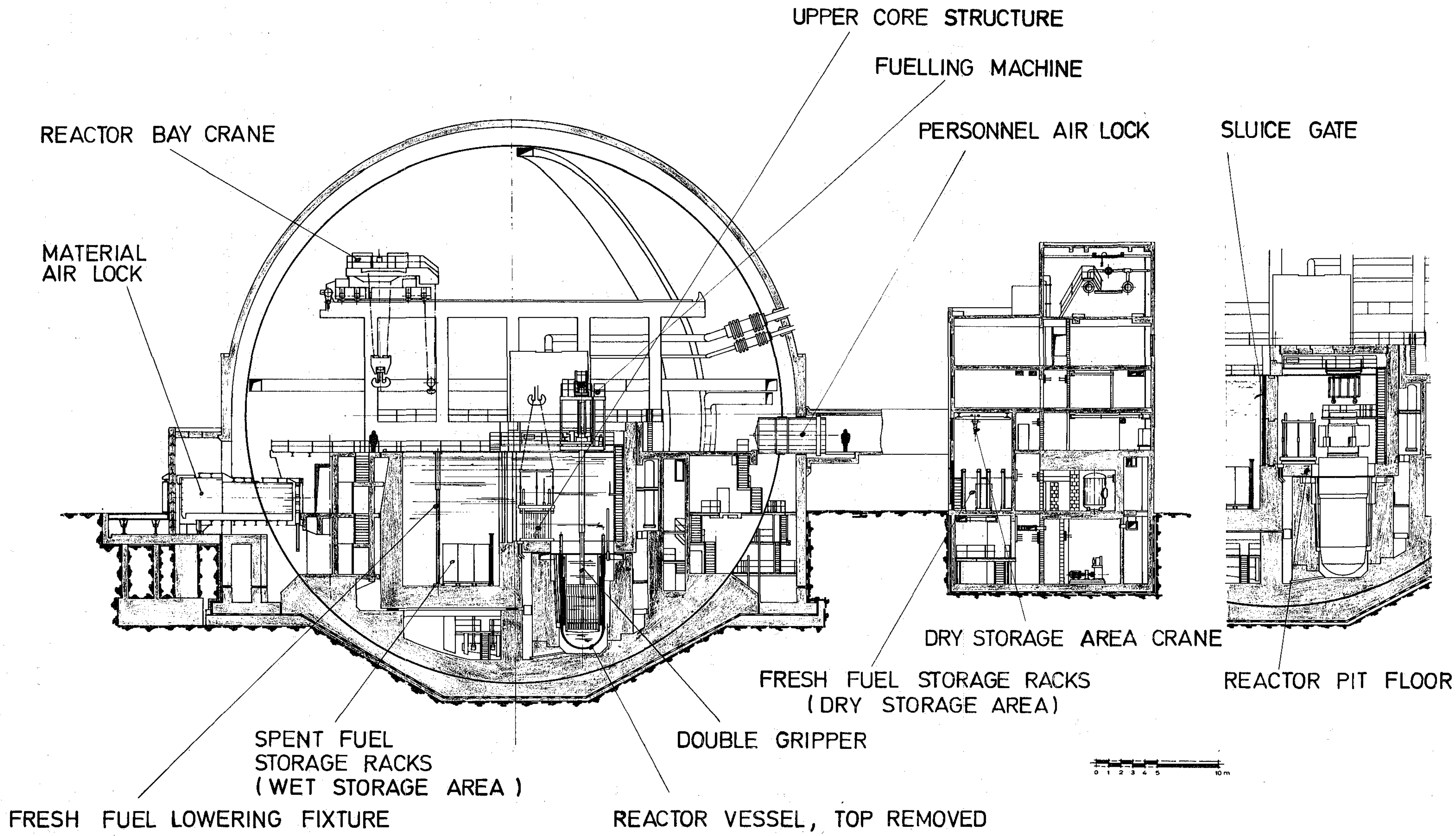
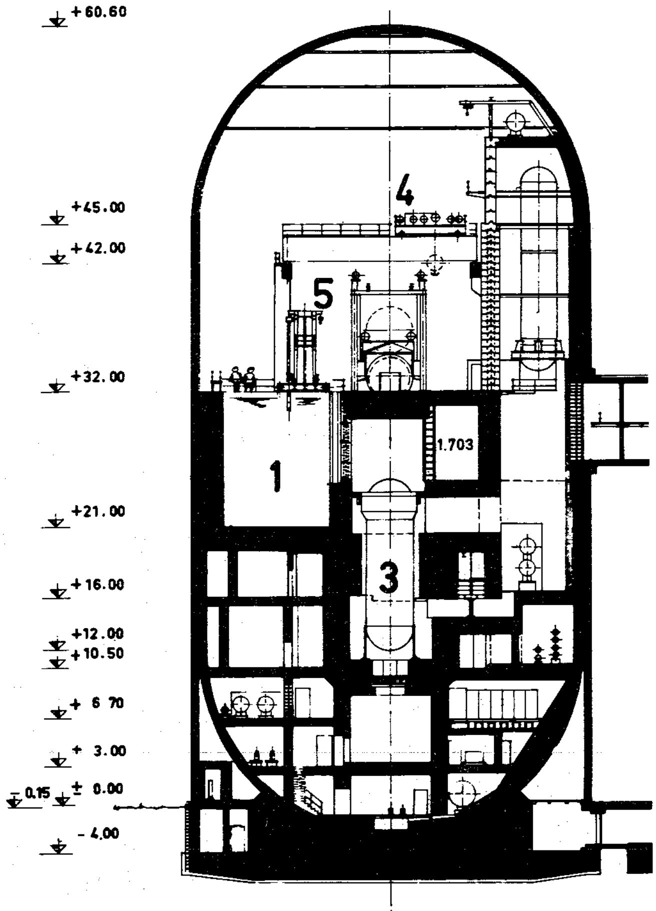
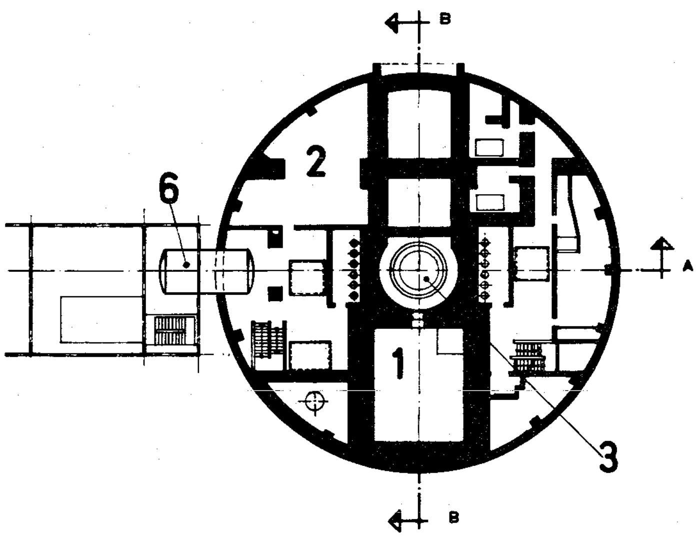


FIG. 2 GENERAL LAY-OUT OF DRY AND WET STORAGE AREAS, REACTOR BAY AND FUEL TRANSFER MECHANISMS FOR KWO POWER STATION



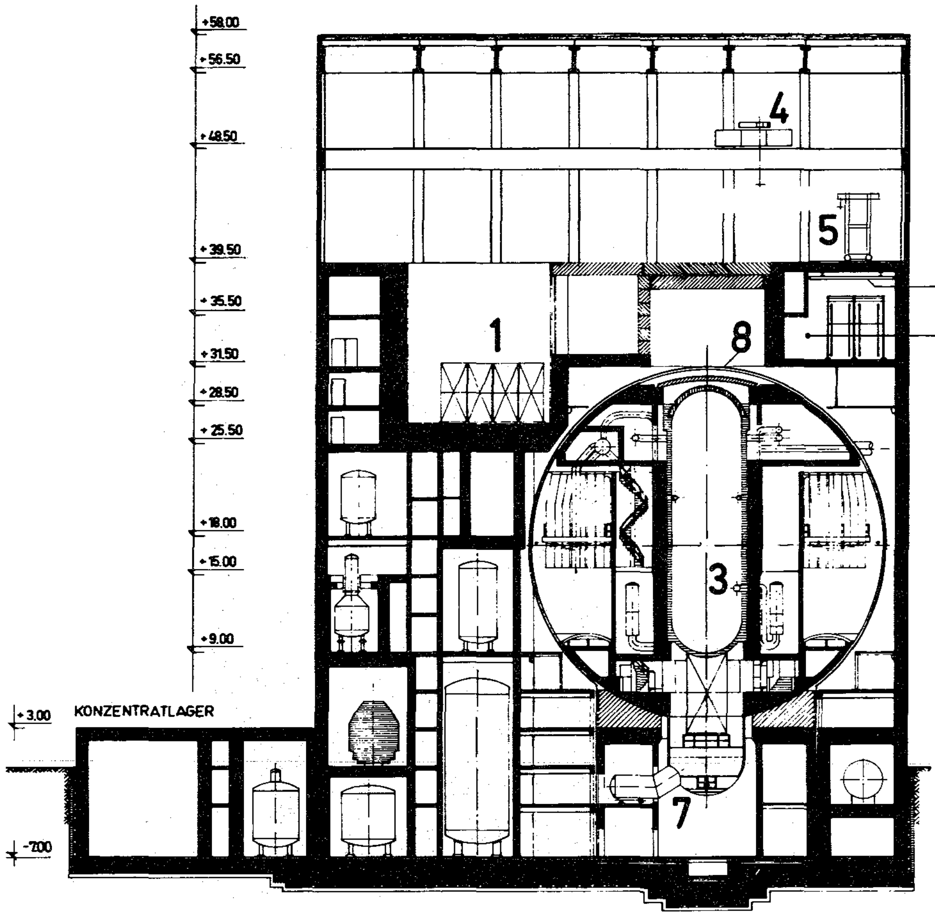
SCHNITT B-B



GRUNDRISS + 25.00

1 wet storage
2 dry storage
3 reactor vessel
4 main crane
5 fuelling machine
6 material lock

Fig. 3 General layout of dry and wet storage areas, reactor bay and fuel transfer mechanism for KWL reactor.



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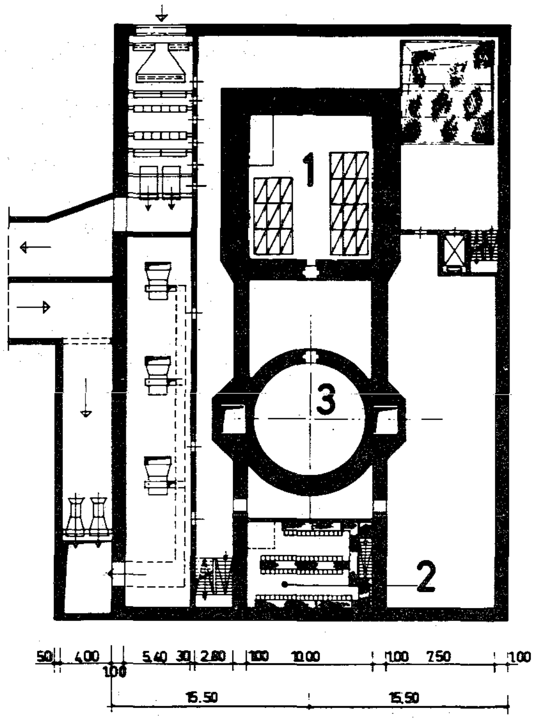

GRUNDRISS +35.50 m

1	wet storage
2	dry storage
3	reactor vessel
4	main crane
5	fuelling machine
7	personnel lock
8	containment cover
9	dry storage crane

Fig. 4 General layout of dry and wet storage areas, reactor bay and fuel transfer mechanism for KWW reactor.

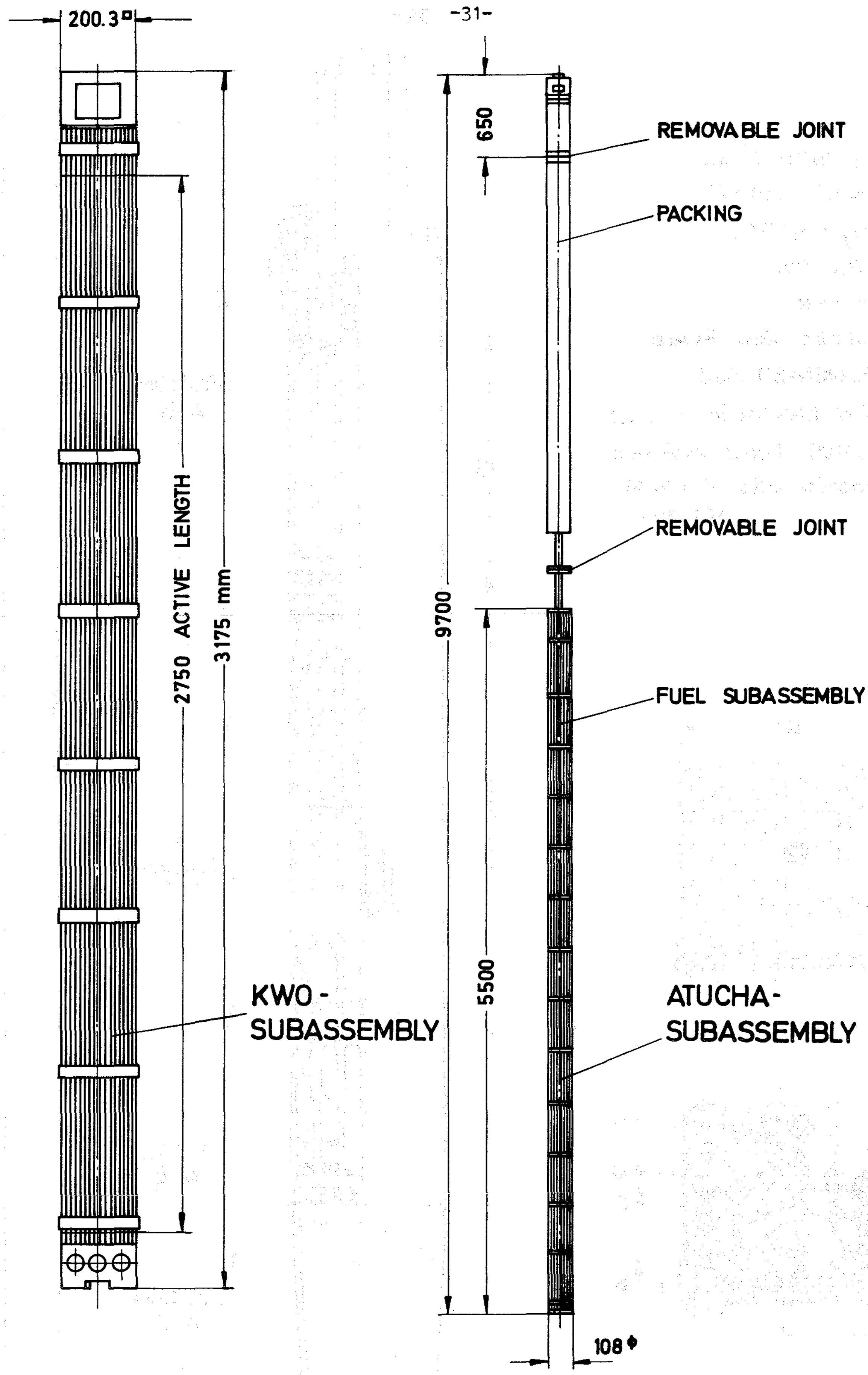

FIG. 5 FUEL SUBASSEMBLIES FOR ATUCHA AND KWO POWER STATIONS

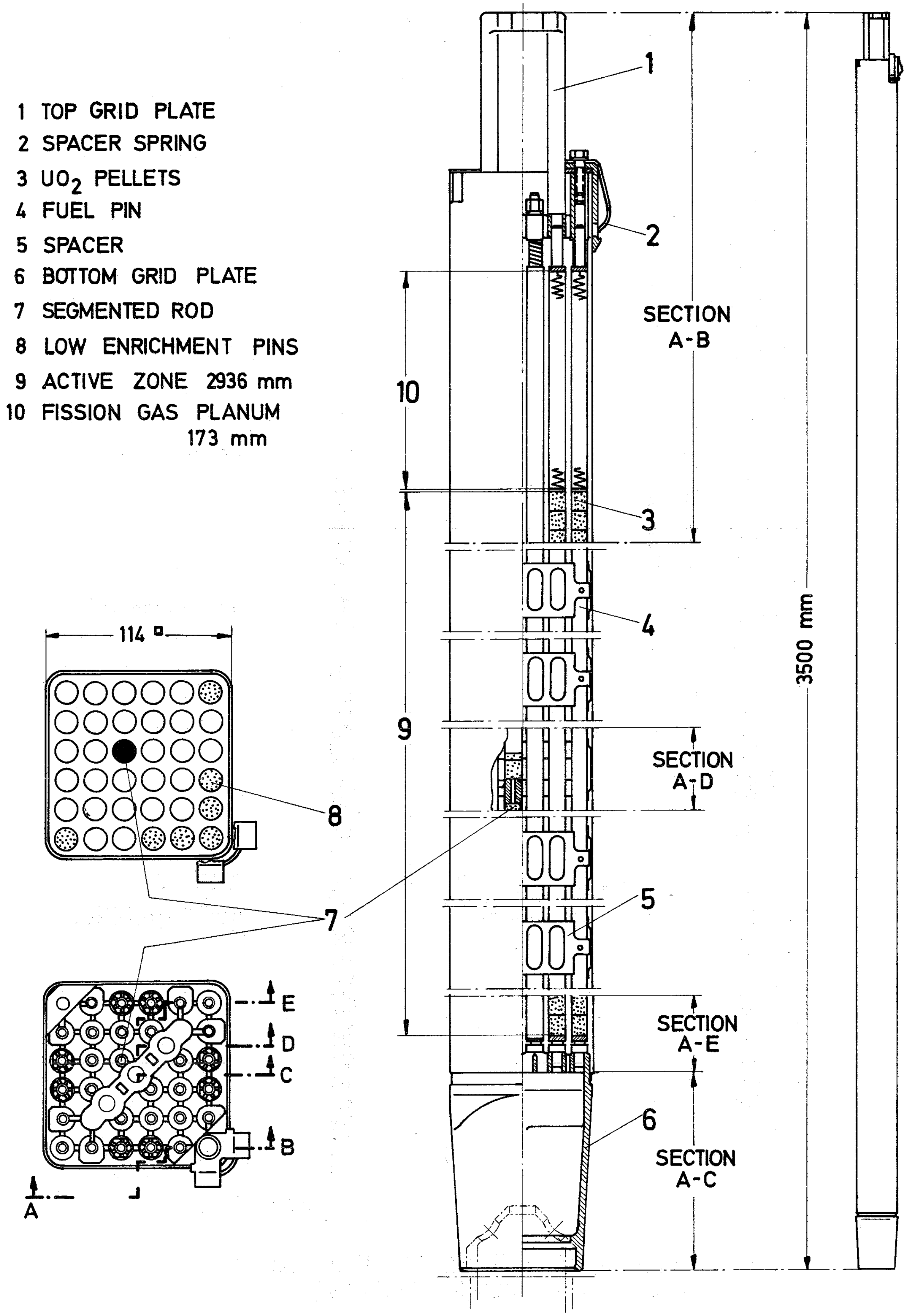

FIG. 6 FUEL SUBASSEMBLY FOR KWL POWER STATION

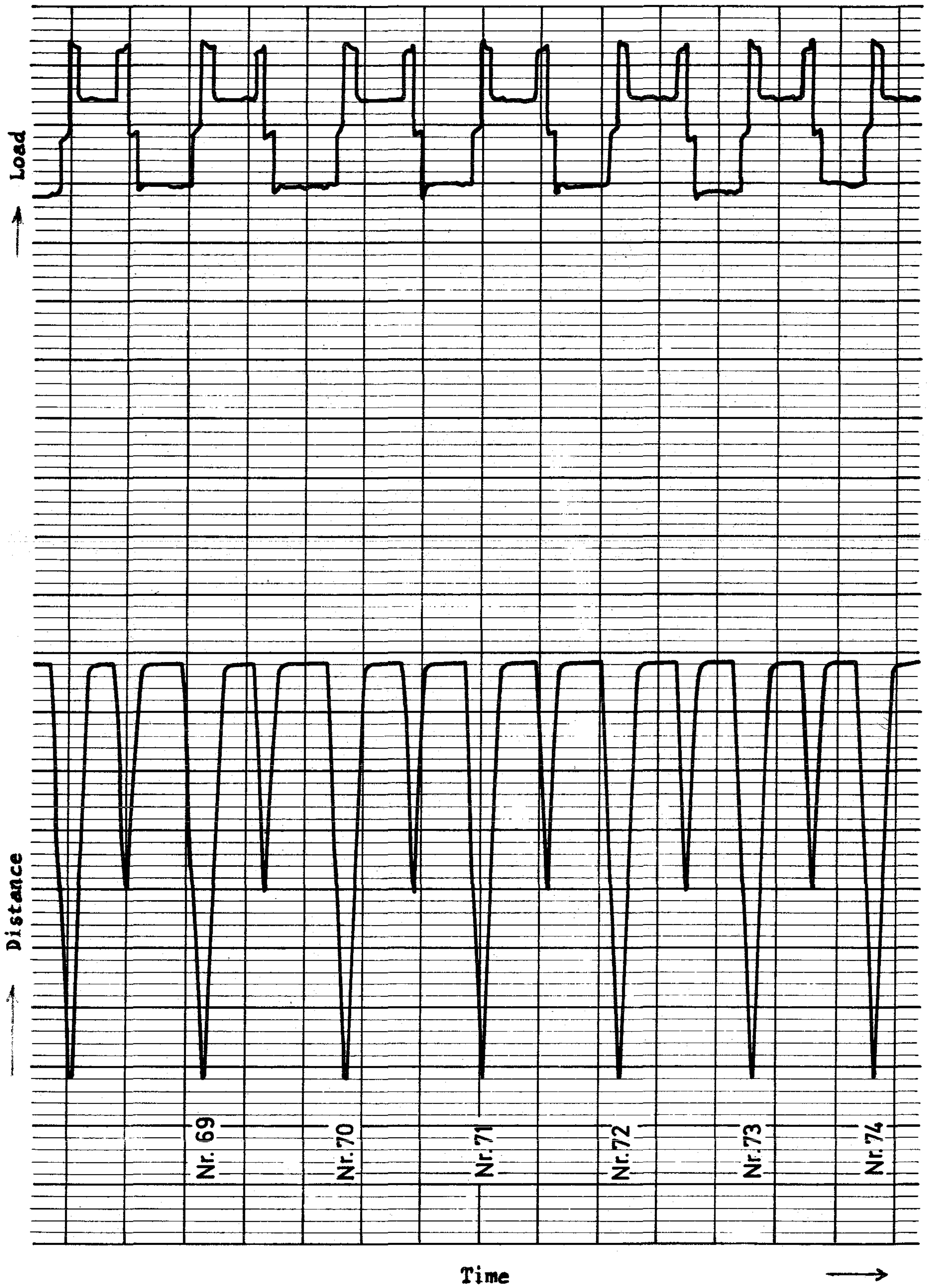

FIG. 7a ORGINAL PRINT - OUT FROM DISTANCE - LOAD REGISTRATION INSTRUMENT DURING LOADING OF KWO CORE

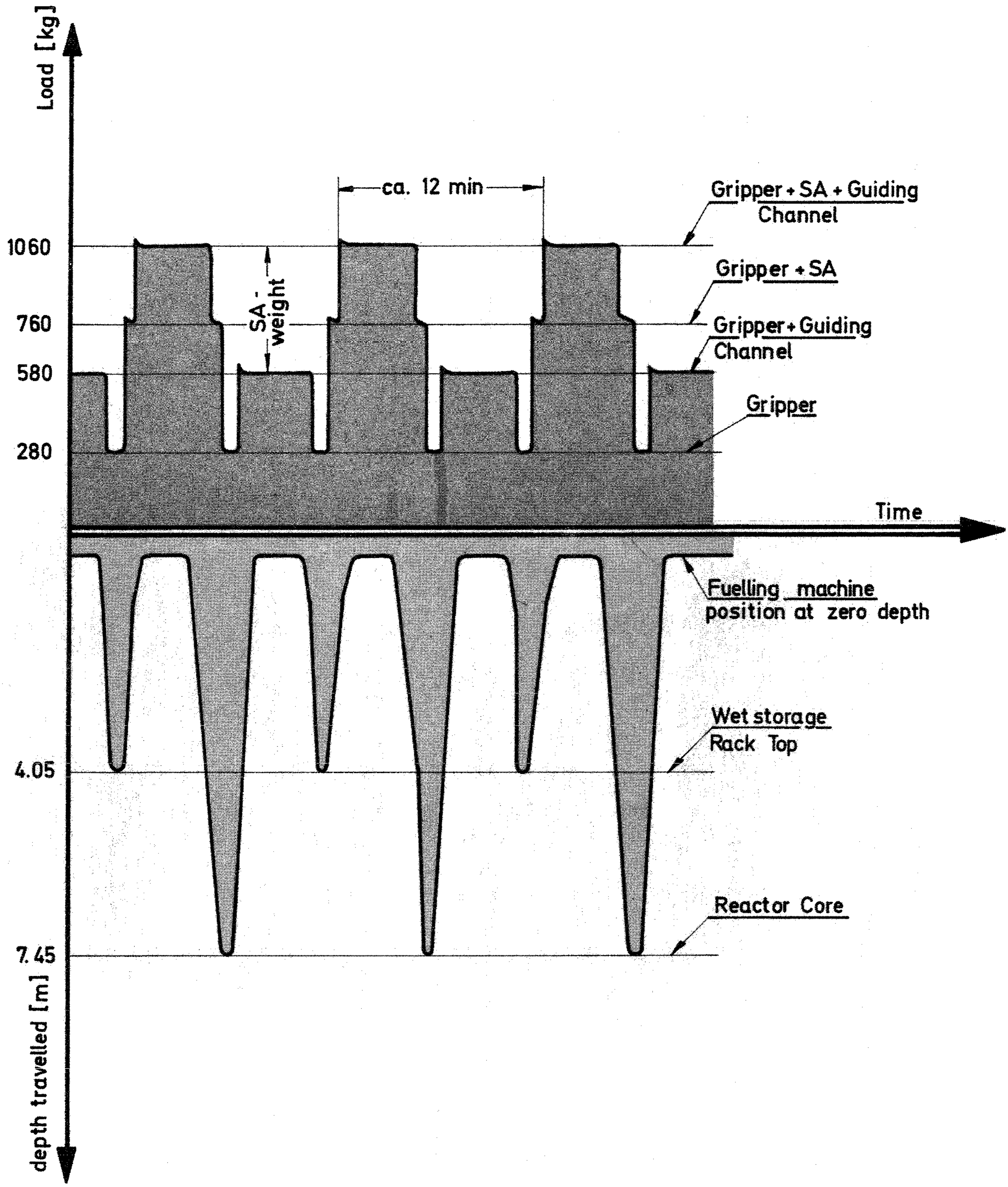

FIG. 7 b DISTANCE-LOAD DIAGRAM FOR FUELLING MACHINE DURING LOADING OF KWO REACTOR CORE

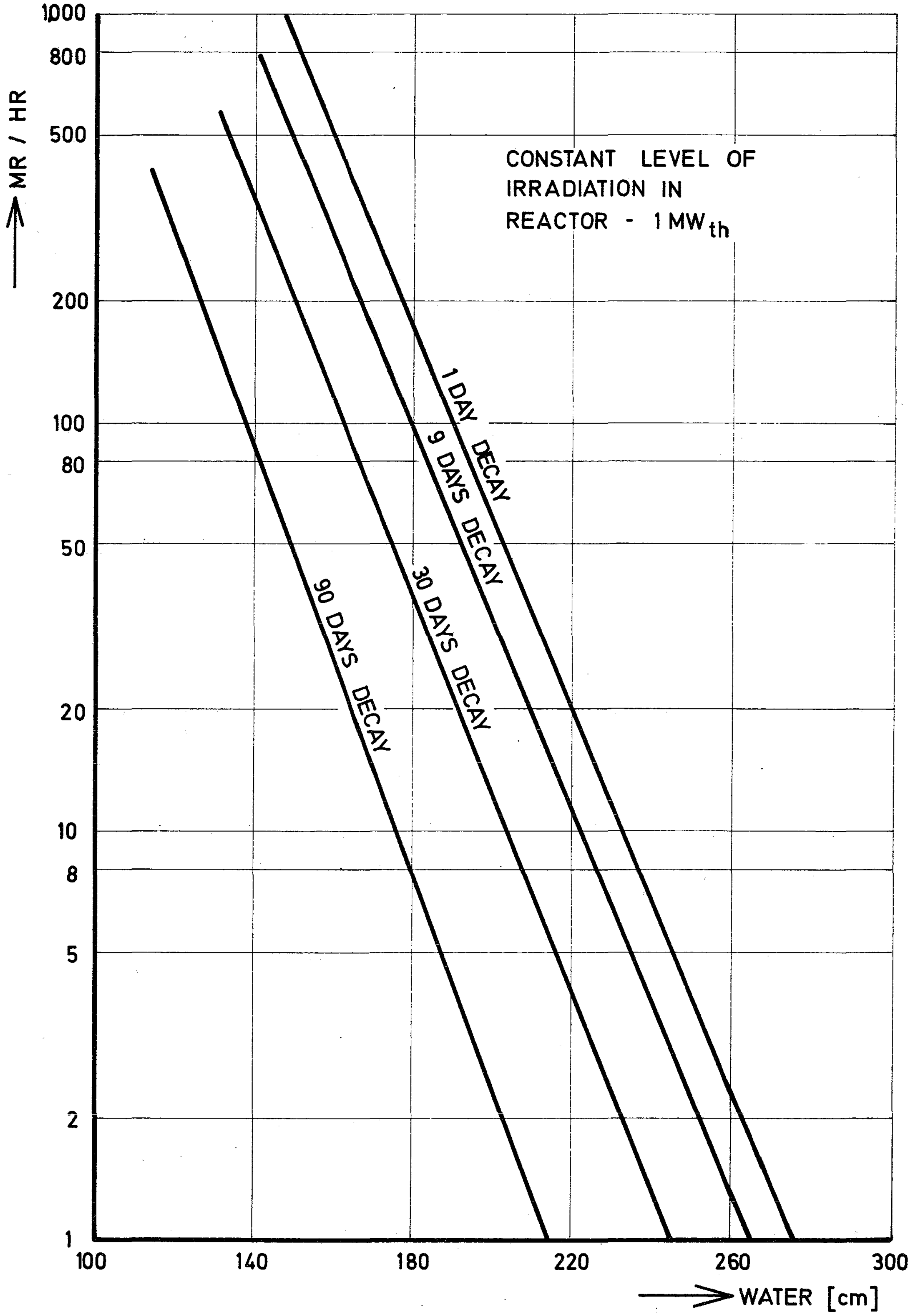

FIG. 8 ACTIVITY OF AN IRRADIATED SUBASSEMBLY AS MEASURED BY A DETECTOR AT DIFFERENT UNDER WATER DISTANCES, WITH DECAY TIME AS PARAMETER.

Appendix I

Determination of fissile material content in the discharged subassemblies from a boiling water reactor

S. Kornbichler

1. Introduction

The determination of the total amount of fissile isotopes in a discharged batch of subassemblies can be devided in two steps; firstly, the determination of fission and absorption rates as functions of space and time - i.e. practically of the space-time-dependent isotopic concentrations, and secondly the integration of the space-dependent isotopic concentrations over the total volume of a discharge batch.

The second step, which sounds quite trivial, is a very important one with respect to accuracy. Large errors in calculated local isotopic concentrations may be averaged out to a fairly low degree by integrating over a large volume.

The calculation of the space-time-dependent isotopic concentrations has to be done consistent with space-time-dependent power and flux distribution in discrete time steps. Therefore determination of isotopic concentrations requires a complete description of the reactor history with respect to coolant conditions, control rod sequence, refueling patterns and overall thermal power generation.

2. Method of calculation

Because of the complicated geometric distribution of material properties within a boiling water reactor there is no possibility of doing an "exact" calculation of isotopic concentrations. Instead of this, only an approximate solution of the problem is possible and is usually carried out in the following manner:

a) The reactor is subdivided into rectangular cubes. In the horizontal direction the cross section of a cube corresponds to a so called "bundle cell". A bundle cell contains the fuel bundle, the half width of the adjoining water gap and, if existent, the half width of the two wings of an adjoining control rod.

   The principle of calculation is to replace the actual heterogeneous cubes by homogeneous ones, using adequatly averaged cross sections, and to solve the neutron diffusion equation for the whole reactor made up of homogeneous cubes.

b) The properties of a particular cube are calculated in advance as a function of <u>average</u> moderator void content and <u>average</u> burn up within the cube, assuming that there is no mutual interaction between different cubes. In this way, a kind of library can be established for the effective homogenized neutron cross sections for all types of cubes which ever may come up in the burn up history of a reactor core.

   Naturally, this library can only give an <u>approximate</u> description of the real properties of the cubes, as the details of local burn up history within the cube are not considered.

   Nevertheless, this model is the best one which presently can be used for reactor design, as it keeps costs for computer time within reasonable limits.

c) Once the effective homogenized cross sections of the cubes as a function of voids and presence of control rods are given, the spatial flux and power distribution within the reactor for a particular control rod pattern, overall coolant flow and burn up distribution can be determined. This evaluation of power distribution is based on both calculational and instrumentational means. Boiling water reactors of the presently known commercial type, are equiped with incore neutron flux monitors which show axial thermal flux distributions at about 20 radial positions of the core. These flux measurements together with two dimensional core diffusion calculations for the horizontal core cross section give the macroscopic power distribution.

d) Integration of local power between two time steps over the enclosed time interval yields the differential local increase in burn up. Following the history of a cube throughout its irradiation time in the core, the integrated burn up at the time of discharge of the bundle containing the cube can be given.

e) Finally, from the isotopic library mentioned in item b), the average isotopic concentrations within the cube can be taken as a function of the burn up of the cube and the average moderator void content which the cube experienced during its irradiation history. Again it must be noted, that no details of isotopic distribution within the cubes are given.

3. Input data and calculational models

To make all the calculational steps mentioned above, the knowledge of a detailed geometric and material description of the core, comprehensive operational data and adequate physical models programmed on a digital computer are required. The essential necessary information is as follows:

a) For the determination of the initial properties of a reactor one needs:

(i) Geometrie design of fuel rods, fuel bundles, absorber elements and the core arrangement

(ii) Material specifications of all structural materials

(iii) Fuel enrichment

(iv) Amount and spacial distribution of black and of burnable

(v) Neutron poisons

b) For following the burn up history of the installed fuel one needs for discrete time steps:

(i) Thermal reactor power

(ii) Overall coolant flow

(iii) Coolant subcooling

(iv) Coolant pressure

(v) Control rod positions

(vi) Readings of incore neutron flux monitors

(vii) Fuel element reload patterns

c) For determining isotopic concentrations from initial reactor properties and burn up history a set of physical models and corresponding computer codes together with consistent nuclear data and empirical corrections are required. The most important problem with generally known physical models and corresponding computer codes is that they fail to give exact results for each type of reactors. Therefore, for a particular reactor type, the results obtained by the physical models must be fitted to experimental data. This fitting is very sensitive to the nuclear input data used in the model (cross section libraries) and may be applied either to the results of the model or to the input data. So, essentially, one needs:

(i) Physical models and corresponding digital computer codes for neutron energy spectrum, neutron flux distribution and burn up calculations

(ii) Consistent libraries for microscopic isotopic energy dependent cross sections

(iii) Consistent libraries for integrated cross sections (like resonance integrals)

(iv) Consistent empirical correction formulas for macroscopic cross sections (like blackness of absorber materials) and macroscopic flux and power distributions.

4. Accuracy of calculation

It is difficult to estimate the accuracy ofcalculated neutron isotopic concentrations,because large local errors may be averaged out by integration over large volumes. Also detailed comparisons between calculated and measured mean isotopic concentrations of a boiling water reactor discharge batch have not been performed sofar.

Comparisons of measured local Plutonium concentrations in discharge fuel of the "Versuchsatomkraftwerk Kahl" (VAK) with calculated numbers are shown in the following table:

Isotope	relative error of calculation		
U 235	0 %	to	- 10 %
Pu 239	-5 %	to	- 15 %
Pu 240	-15 %	to	- 25 %
Pu 241	+30 %	to	+ 40 %

10 samples from 10 different locations in two different fuel rods were taken. The spread of the deviations between calculated and measured numbers is shown in the table.

The spread of the deviations for one isotope may be reduced by an order of magnitude when averaging over the whole discharge batch; but the magnitude of the deviations and the unchanged sign for every isotope indicate a systematic error in the calculation of siotopic concentrations as a function of burn up.

At the present time it seems that the average burn up of a discharge batch may be calculated with an accuracy of some 2 % or 3 %, provided a process computer is available. But large errors of the order of 10 % or more are be expected in case the isotopic concentrations are calculated from relatively exact burn up data.

5. Conclusions

The determination of the fissile material content in the discharged batch of subassemblies from a boiling water reactor is cumbersome and expensive. At present the state of the art in calculating isotopic concentrations does not guarantee accuracies better than 10 %.

Improvements may be gained in the future, if numerous systematic comparisons between measured and calculated isotopic concentrations of discharge batches are made and if the results are fed back to the calculational models. However, because such empirical adjustments are based on integral measurements, the range of validity will always be very small. Therefore, for each reactor type, these adjustments have to be made separately, which is a very time and cost consuming procedure.