

NRC Translation #458

9/78



GRS
Gesellschaft fuer Reaktorsicherheit
(GRS) mbH
(Society for Reactor Safety)

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INVESTIGATIONS ON THE COMPARISON OF
GREATEST POSSIBLE ACCIDENT CONSEQUENCES
IN A REPROCESSING PLANT AND IN A NUCLEAR
POWER PLANT
(CRITICAL COMMENTS ON WORK REPORT AB-290)

[Stamp: 15 June 1978]

GRS - A - 59 (NOVEMBER 1977)

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SUMMARY

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Work Report 290 of the Institute for Reactor Safety of the Technical Supervision Union e.V. (IRS) showed that, in the event of a postulated disregard of all safety installations planned for the Decommissioning Center (NEZ) or present in the nuclear power plant, the danger potential for nuclear power plants and for the Decommissioning Center is approximately the same. This gives rise to the conclusion that the safety installations for a nuclear decommissioning center, more particularly the afterheat discharge system, must satisfy comparable criteria as for a nuclear power plant. It was expressly determined in the report that the numerical data given on the radiological loads represent no risk statement under realistic conditions and that an evaluation as absolute statement is inadmissible. In the same way, the numerical values are taken over uncritically from various sides, published and erroneously interpreted.

The AB 290 contains, as was stated on the title page, only provisional results. The Society for Reactor Safety mbH as successor society of the IRS submits in the following a reworking and detailed explanation of the report whereby the latest state of art and application data are used as the basis:

The design data of the nuclear decommissioning center,

The first results of the German risk study,

The latest data for release, spread and dose factors.

The results of the thermodynamic computations of the extreme improbable cooling breakdown in the fuel element receiving pools and in the high level radioactive waste containers can be summarized as follows:

In order to keep the water level during evaporation to the level required for avoiding a heating up, the following replenishment is required:

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Fuel element receiving pools:
ca. 50 m³/h after 10 days
Waste containers:
ca. 21 m³/h after 2 days.

*Numbers in the right margin indicate pagination in the original text.

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Accordingly, there are either 10 or 2 days available in order to replenish the relatively slight water quantities for preventing melt. These coolant quantities per time unit are so small that they can be even brought by tank car transport. In the same way, an improvised hose connection to a 1½" to 2" line (for example, fire hydrant) is sufficient to supply the required coolant water quantity at the customary pressure level in the supply network.

For this reason, a melt accident is out of the question for the fuel element pool and the HAW containers is out of the question since the possibly required emergency measures are easy to carry out.

A disastrous fission product release and the high radiation exposures connected therewith are, for this reason, impossible. The corresponding tables in Work Report 290 are for this reason not suitable for making statements on the radiological danger.

A nuclear melt accident in the reactor is not considered in the Atomic Legal Licensing Procedure and is not mentioned as an event to be considered in the standard Safety Report ("Arrangement of Items and Organization for a Standard Safety Report for Nuclear Power Plants with Pressurized Water Reactor or Boiling Water Reactor", appearing in the Joint Ministerial Gazette, Edition A, No. 6 of 30 August 1976, pp. 418 ff.) since, as a result of the costly emergency cooling system, the probability of occurrence for such a case is so small that it can practically be excluded. Nevertheless, radiation exposures were computed or taken from technical literature for just such a sequence of events (Chapter 7).

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The own new computations should show, for the sake of example, that the results published in Work Report 290 cannot be used for absolute statements with respect to radiological danger.

The differences with the results of other authors (WASH-1400, KFK 2433) can be explained by the limiting conditions selected. More exact, risk-relevant statements or evidence must be taken from the German "Reactor Safety Study" presently being written.

As has already been explained, a core melt is practically out of the question in a nuclear power plant. For this reason, computations of radiological effects are also not required. If such statements, however, had to be made, the relative numerical values mentioned in AB 290 need not be used but the results achieved in Chapter 7 of the present report. In order to be able to make a statement concerning the risk connected with a nuclear power plant, it is necessary to analyze not only the scope of damage but also the

probability of occurrence. These risk-relevant statements are made by the German risk study presently being worked on.

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1. INTRODUCTION

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The internal Work Report 290 had the task of comparing the danger potential of a large reprocessing plant with that of a nuclear power plant. The goal of this comparison was to determine whether comparable safety installations are necessary for the reprocessing plant as for a nuclear power plant. For this reason, the contracting authority specified that the investigations would be carried out using the limiting condition that no safety systems are either present or effective. Accordingly, the probability of occurrence of such a sequence of events was also not considered but a total cooling breakdown was assumed which always leads to melt and, in the case of the specified limiting conditions, to high radiation exposures. In the case of accident analyses for a nuclear power plant in connection with the Atomic Law Licensing Procedure, a coolant loss accident with subsequent core melt was not considered since the probability of occurrence for such a combination of events was so low that it can be ruled out of the question. In the case of a coolant loss accident, safety systems are used which feed water into the reactor pressure vessel in order to again raise the water level in the pressure vessel and ensure the long-term cooling. The following system functions are involved herewith:

High pressure feed,

Accumulator feed,

The low pressure feed for flooding and subsequent circulation activity.

Depending on the specific accident (large, medium-sized or small leak), the above-described safety installations are placed in operation. As a rule, four subsystems are specified for a required system function whereby the effectiveness of emergency cooling is ensured by the operation of two subsystems. On the basis of this design principle, it is used as an assumption in the licensing procedure on a worldwide basis that these safety systems are sufficient to avoid a core meltdown.

The quantities used in computing the potential radiation exposures, for example release altitude, propagation conditions, etc., had therewith only a model character and were selected from the viewpoint of "comparability". They are not suitable for achieving a realistic estimate of radiological effects.

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Further, for reasons of a simpler comparability, an inconsistency in the investigation was taken into the bargain inasmuch as the thermal engineering computations started from an almost-intact building whereas the computation of the radiological effects started

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for practical purposes from a faulty building. The building structures, as safety containment, were not considered in the computations for AB 290 although they even in a defective state clearly reduce the activity released from the core on the way to the outside by deposits and similar effects.

For the abovementioned reasons, it is accordingly completely inadmissible to value the radiation exposures mentioned in Work Report 290 as an absolute statement or as advisory position on the radiological effects to be expected in accordance with the abovementioned events.

In the meantime, a comprehensive safety report for the planned nuclear decommissioning center is available. The data of the nuclear decommissioning center are essentially differentiated from the fictitious data for a reprocessing plant used as the basis in Work Report 290. In the meantime, there is available for the decommissioning center a joint position of the Reactor Safety Commission (RSK) and the Radiation Protection Commission (SSK) in which both commissions make positive statements on the safety-engineering feasibility.

The following-described events show that a meltdown accident is out of the question in the case of high active waste containers (HAW) and in the case of fuel element receiving pools since sufficient length of time is available to feed in the relatively slight water quantities required for control. For this reason, it is unnecessary to compute accident doses for a "meltdown situation".

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The core meltdown accident in the nuclear power plant with pressurized water reactors (DWR) is to be ruled out of the question as has already been explained. Nevertheless, with respect to potential radiological consequences, no recent results are reported which take into consideration computations of other authors in order to avoid a further misuse of the numerical values intended to be relative which were mentioned in AB 290 and to indicate essential differentiations such as short-term and long-term doses.

2. LIMITING CONDITIONS IN WORK REPORT 290

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The following limiting conditions were specified by the contracting authority:

The initial assumption is to be total breakdown of all cooling and ventilation systems.

The release of radioactive substances from the meltdown results without reduction mechanisms directly into the environment.

The release takes place near the ground.

Additionally, the following conservative measures are taken which led compulsorily to a further estimate of potential effects:

An infinite irradiation time was assumed for the fuel elements stored in the fuel element storage.

The heat release by convection and the heat accumulator capacity of the building were disregarded.

The heating up of fuel elements starts after evaporating up to fuel element upper edge.

The release factors were assumed conservatively. The release duration was negligibly short.

The propagation was computed conservatively with the parameters of Pasquill for stable and additionally neutral weather conditions.

Increased dilution owing to swirling in the building at the time of release, by deposit during transport into the atmosphere as well as by fluctuations in wind direction was not taken into consideration.

The most unfavorable values were always used for the dose factors. These involved rather old data in which the most recent information was not yet taken into consideration.

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3. INVESTIGATIVE SCOPE

The conservative factors used as a basis in accordance with the contract in Work Report 290 or for reasons of simple comparability are comprehensively explained.

In order to augment the discussions in Work Report 290, it is described which systems and components have to be assumed as broken down in spite of reliable design before there can occur any heating up in the fuel element or waste container.

For the postulated event (total breakdown of cooling), using the now well-known data from the decommissioning center, the time sequences for the heating up process and the cooling water quantities required for preventing meltdown are redetermined.

On the basis of these time and quantity data, a description is given of the possibilities available for reliably preventing meltdown accidents in the fuel element receiving pools and in the HAW container.

In order to arrive at a technical evaluation of the sequences of events considered in Work Report 290, it was sought on the basis of statements made in the Safety Report concerning the plant statuses and design data to arrive at a qualitative statement concerning the danger connected with these sequences of events.

For the postulated nuclear meltdown accident with the pressurized water reactor, the latest results are presented although this event combination, as was explained in detail in Chapter 1, is quite improbable and, for this reason, is not considered in the licensing procedure.

4. SHORT DESCRIPTION OF COOLANT CIRCUITS

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The fuel element storage pools in the HAW containers are fitted with cooling systems for removing afterheat.

In the case of the fuel element pool, pool water is pumped to the heat exchanger using a circulating pump. The heat exchanger discharges heat to the intermediate coolant circuit. From the intermediate coolant circuit, the heat finally arrives at the main cooling water system through a second heat exchanger. The main cooling water is re-cooled through a wet cooling tower. Replacement water is removed from a self-contained cooling pond. The coolant circuits are designed as 3 x 100% circulations. In (2 out of 3) operating mode, the pool temperature is kept, according to the Safety Report, to ca. 40° C. In the case of a (1 out of 3) mode of operation, this temperature rises from 40 to 60° C.

The standard power supply for the cooling installations takes place through two separate independent feedins. Both consist of overhead wires which are supplied by separate nodal points of the plant. In the event of breakdown of connection 1, the decommissioning center is further supplied by connection 2. Only in the case of the quite improbable simultaneous breakdown of the two independent feedin systems would a situation arise where emergency power drop occurs.

In order to supply the safety-engineering important consumers, a diesel emergency power plant is specified which is divided consistent with the process engineering concept into 3 x 100% emergency cooling loops. Each loop is supplied by two diesels with 50%.

The cooling installation of the HAW containers is built in the same way. Nevertheless, a coolant circuit is lacking. The intermediate coolant circuit takes the heat directly from the container content to be cooled using an oversized cooling loop. According to the Safety Report, the heat discharge installations correspond to the single error criterion.

5. TIME LAPSES IN THE ASSUMED COOLING BREAKDOWN IN THE FUEL ELEMENT RECEIVING POOL AND THE HAW CONTAINER OF THE DECOMMISSIONING CENTER

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The fictitious plant dealt with in Work Report 290 differs essentially from the data given for the decommissioning center in the Safety Report. The following table contains a number of the differences decisive for the heating-up process.

Fuel Element Storage Pool

	AB 290	Decommissioning Center
Number of stored fuel elements	1400	6544
Decay time of fuel elements [d]	200	485
Afterheat capacity [MW]	25.4 (infinite irradiation time)	31.0
Water volume in total pool to be considered [m ³]	12038	13806
Total pool cross section [m ²]	842 (with subtraction of discharge pool)	1062
Difference in height available for evaporation [m]	10 (evaporation up to core upper edge)	10.5 (evaporator up to core center)

Waste Container

Specific afterheat capacity of waste container [W/l]	16	12.8
Water volume in total pool to be considered [m ³]	839	839*
Total pool cross section [m ²]	256	256*

*These data refer to the waste containers with maximum heat capacity.

Disturbances in the cooling processes of individual fuel element pools can be coped with by shifts from fuel element pools with disturbed cooling into pools with intact cooling whereby insulation can be carried out through compartmentalizing. Further, replenishment of water is made with operational means. Herewith, the temperature is kept at standard operating values or limited to 100° C. The release of radionuclides from the fuel into the coolant is insignificant in the temperature available. Nevertheless, in order to obtain an impression concerning the time sequence in the case of an assumed and not improbable cooling disturbance, there results with a disregard of the above countermeasure, using the actual data for a decommissioning center, the following picture for the fuel element pool:

Heating up to evaporation temperature	ca. 1.3 d
Evaporation of coolant inventory	ca. 9.3 d
Adiabatic heating up to meltdown start (1700° C)	ca. 1.0 d
Heating up to meltdown start taking into consideration the storage capability of concrete	ca. 2.1 d

These time data were determined taking into consideration a time-limited irradiation period (three years). The convective heat transport through a ventilation channel to the outside (in the case of nonfunctioning ventilation) was not considered since it would only affect meltdown start on a nonessential basis. An intact ventilation system cannot be assumed in the heating up phase since, with increasing temperatures, damages cannot be avoided on the whole system (seals, flexible connections, thermal expansion). Furthermore, the capacity of the intact ventilation system cannot prevent a meltdown but only lengthen time up to start of meltdown.

The heat engineering computation provides for waste containers the following situation and even here possible countermeasures such as, for example, pumping out into intact containers were disregarded:

Heating up of concentrate from 60 to 100° C	ca. 0.13 d
Evaporation of the concentrate	ca. 1.7 d
Heating and evaporating the cooling jacket contents	ca. 0.12 d
Adiabatic heating up to meltdown start (1000° C)	ca. 0.22 d

Also in the case of the waste container, a meltdown (residual salts) is only conceivable in the case of a total breakdown of cooling without consideration given countermeasures.

The scavenger air system is assumed to be intact in the heating up phase although it has no significance with respect to heat removal.

In order to maintain the water level during evaporation to the level required for avoiding a heating up, the following feedin is required:

Fuel element receiving pool:
ca. 50 m³/h after 10 days

Waste containers:
ca. 21 m³/h after 2 days

There are accordingly 10 or 2 days available as the case may be in order to feed in the relatively small quantities of water for preventing meltdown.

For this reason, a meltdown accident can be considered out of the question for the fuel element pool and the HAW containers since the emergency measures possibly needed are easy to accomplish.

This finding is confirmed by the accident considerations described below (Chapter 6).

Within the scope of the licensing procedure for nuclear power plants, coolant loss accidents were investigated in detail. In the German risk study, the course of the accident in the case of an additionally assumed but improbable breakdown of emergency cooling is further analyzed. For this reason, it is necessary to go further into this series of problems here. Insofar as the decommissioning center and, more particularly, the fuel element pool and waste containers are concerned, a number of parameters should be stated which characterize the progress in time of an assumed disturbance.

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	NUCLEAR DECOMMISSIONING CENTER	
	Fuel Element Storage Pool	Concentrate Container
System pressure and temperature	No pressure; low temperature level ca. 1/bar/; ca. 50° C	No pressure, low temperature level 1/bar/; ca. 50° C
Time for instituting counter-measures	During the first 11 days after failure of any aftercooling installations, constant temperature level at ca. 100° C	During the first ca. 2 days after failure of any aftercooling installations, constant temperature level with ca. 100° C
Afterheat	In the range of 1 to 2 days, almost constant = ca. 31 MW	In the range of 1 to 2 days, almost constant = ca. 13 MW*
Required feedin rates and possible improvisational measures	11 days after breakdown of pool cooling, the following feedin quantity is required as replacement for the evaporation (in the time period up to 11 days, no feedin needed) ca. 50 $\frac{m^3}{h}$	Ca. 2 days after failure of cooling, the following feedin quantity is required as replacement for evaporation (in the time period up to ca. 2 days, no feedin is required) ca. 21 $\frac{m^3}{h}$ *

*These data refer to the HAW containers with the greatest heat efficiency.

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NUCLEAR DECOMMISSIONING CENTER

Required feedin rate and possible improvisational measures (continuation)	Fuel Element Storage Pool	Concentrate Container
	<p>With total breakdown of cooling, there is a wide spectrum of measures to be improvised which lead to control and thereby prevention of the meltdown process since:</p> <ol style="list-style-type: none"> Available time interval is so great that measures are only required after days (ca. 10 days). Coolant quantities per unit of time are so small that this quantity can even be supplied by tank car transport. Likewise, an improvised hose connection is sufficient to a 1½" to 2" line (for example, fire hydrant) to supply the required coolant quantity with the customary pressure level in the supply network. 	<p>With total breakdown of cooling, there result similar conditions as with the fuel element storage pool. They are differentiated only in the time interval and in the feedin rates.</p> <ol style="list-style-type: none"> Time interval for improvisational measures leading to control and thereby prevention of meltdown process, ca. 2 days. Feedin rate for coolant only half of the required coolant quantity for the fuel element storage pool.
Retrofit possibilities	Increase of redundancy of the fuel element pool. Cooling is no problem and feasible without great effort at any time.	Like fuel element storage pool.

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NUCLEAR DECOMMISSIONING CENTER		
	Fuel Element Storage Pool	Concentrate Container
Cooling water supplies	<p>Water storage quantity 10^6 m^3 in a pond on the operating terrain. Within a time range up to ca. 10 days, an evaporation cooling of the fuel element can be maintained by institution of improvisational measures over a very long time (up to ca. 800 days).</p> <p>When heating the water storage quantity (10^6 m^3) from a maximum initial temperature of 35°C to 95°C, i.e., without evaporating, there results the possibility of keeping the fuel element to a temperature level of ca. 100°C over a time period of ca. 100 d.</p>	<p>Water storage quantity 10^6 m^3 in the same pond as for the fuel element storage pool on the operating terrain. Within a time period up to ca. 1.5 to 2 days, an evaporation cooling of the bottom settlings (fission products + salts) of the concentrate container can be maintained by institution of improvisational measures over a very long time (800 days).*</p> <p>The same situation results as with the fuel element storage pool, and there always result as a consequence of lesser afterheat longer time periods for the cooling of ca. 230 d. (Afterheat assumed constant.) *</p>

*These data refer to the HAW containers with the greatest heat efficiency.

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6. ACCIDENT CONSIDERATIONS

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The operating and emergency cooling installations for the fuel element storage pool and the HAW containers specified in the nuclear decommissioning center are designed such that, with the help of one of the always-three cooling loops available, the cooling without evaporating of water from fuel element pools or HAW containers can be maintained.

The short description (Chapter 4) shows that the emergency cooling installations of the nuclear decommissioning center correspond to the state of safety technology in the case of modern power reactors. Since with respect to the power reactors for control of cooling accidents in the nuclear decommissioning center always more time is available (cf. Chapter 5), a mere contrast of the reliability of safety systems in nuclear power plants and in the nuclear decommissioning center is only advisable when the time factor is given value. The following are a number of examples in this regard.

Emergency Power Case Fuel Element Storage Pool

The time period available for reestablishing the energy supply amounts to 10 d. There are no probability data available for a network shutdown over this time. A network shutdown during the entire time in question was not observed. Even the shutdown of five emergency power diesel sets of the type that cannot establish over 10 days the necessary (2 out of 6) redundancy is to be characterized as extremely improbable since the average repair time for a diesel amounts to 20 h according to experience. A nonavailability less than 10^{-8} is to be anticipated.

Emergency Power Case HAW Containers

The period available for reestablishing the energy supply amounts to ca. 2 days. Also for this order of magnitude, no reliable probability can be given for a large area network breakdown. The non-availability of the power supply coming from the official network and the emergency power diesels is, as an estimate, 10^{-8} .

Aircraft Crash

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An assumed aircraft crash on the cooling towers would put normal cooling out of operation. Since, however, the cooling of fuel element storage pools and HAW containers can be additionally maintained by a cooling pond, this accident has no effect at all on the coolability. The fuel element pool structure, the reprocessing building and the emergency power diesel building with all associated connecting lines should be protected against aircraft crash and effects of debris.

Breakdown of Individual Coolant Circuits

These accidents have no effect at all on the safety of the plant since each one of the three coolant circuits can maintain the required cooling, i.e., a limiting temperature of 60° can be kept in the fuel element storage pool or in the HAW container even with operation of only one circuit. Only the breakdown of all main coolant circuits leads to exceeding the limiting temperature of 60° C.

Breakdown of All Coolant Circuits and Failure of All Repair and Emergency Measures

Only the hypothetical case that, following a simultaneous total breakdown of all coolant circuits, the repair of the broken-down components within the time available also failed and in addition the emergency measures such as, for example, the setting up of mobile pumps and water feedin from the available water reserve through fire hoses would fail, could lead to heating up the stored fuel elements and the high level radioactive liquid waste. The probability of these hypothetical accidents is so low that a numerical statement is unreasonable. The accident is, for this reason, ruled out of the question.

7. RADIOLOGICAL EFFECTS

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On the basis of available investigations, a meltdown accident can be ruled out for the fuel element receiving pool and the HAW containers of the nuclear decommissioning center. There is practically no situation conceivable in which it is not possible in the time available (10 or 2 d) to guarantee the needed water feed (50 or 21 m³/h). For this reason, it is not reasonable to compute radiological effects for this. The radiation exposures for the fuel element pool and the HAW containers mentioned in Work Report 290 are for this reason groundless with respect to an actual danger. They were used at that time always for purposes of comparison in order to be able to derive requirements for safety installations of the nuclear decommissioning center.

The following depicts rather new findings for a nuclear meltdown accident for a pressurized water reactor. These findings -- publications of other authors and our own computations -- should prevent a further misuse of the relatively intended numerical values mentioned in Work Report 290 and indicate necessary differentiations such as short-term and long-term dose. The computations used as a basis always the extremely improbable event combination of coolant loss accident with subsequent core meltdown and early containment failure leading to releases which, in the American risk study (WASH-1400), are classified in release category PWR 2. PWR 2 has

in the group of nuclear meltdown accidents according to results of the American study a low probability of occurrence with simultaneously the greatest radiological effects. The corresponding accident sequence (coolant loss with failure of emergency cooling and containment failure) leads according to results of the German risk study available so far in the German plant to a later overpressure failure and thereby to essentially lower releases. To the extent that releases in the quantity of the American category PWR 2 can appear from other accident sequences, this is still being investigated at the present time in connection with the German risk study.

According to WASH-1400, Volume VI, Figure VI 13-7 (p. 13-9):

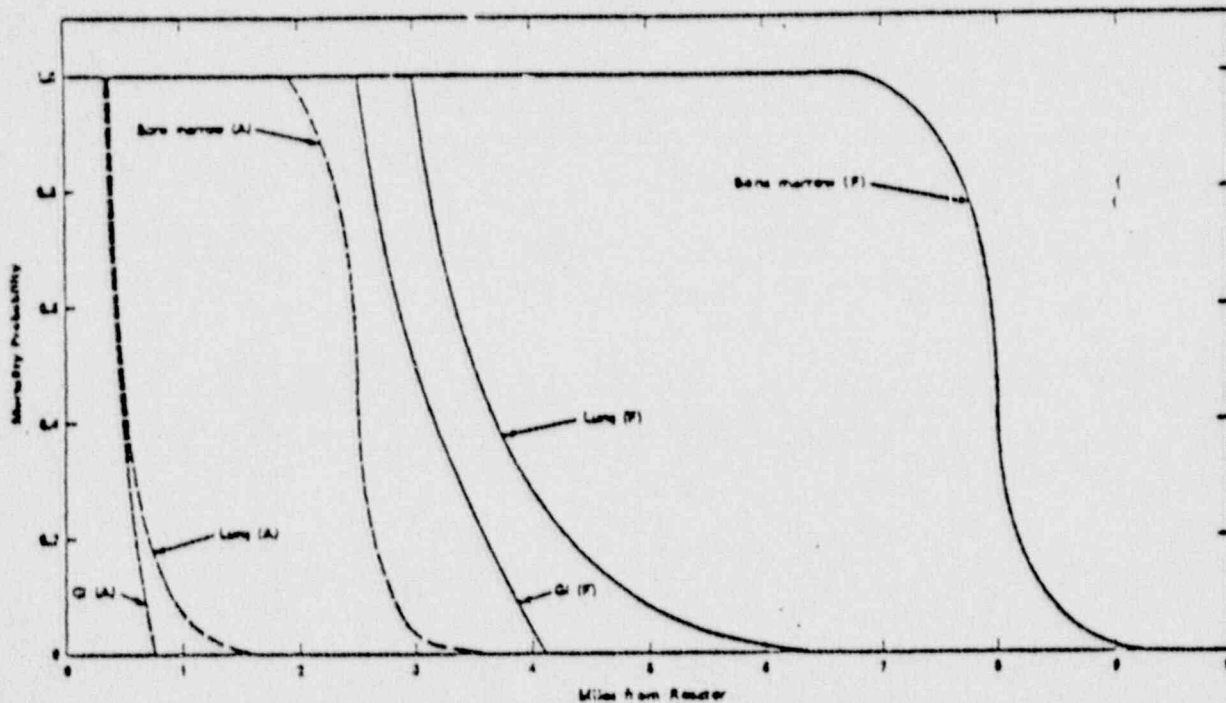


Figure VI 13-7. Mortality probability for an affected population versus distance from reactor for two hypothetical weathers: stability category A, wind speed = 0.5 m/sec; stability category F, wind speed = 2.0 m/sec.

The American risk study reveals that, in the most unfavorable case (weather conditions F, release near ground), radiological effects which can lead in a short time to the death of the affected persons are not to be excluded up to distances between 7 and 9 miles (12 to 15 km). In the case of the most favorable conditions (weather class A), the limiting radius is about 3.5 miles (5 km).

According to the accident consequence model in the German "Reactor Risk Study", KFK 2433, Annual Report 1976 (Department for Radiation Protection and Safety, May 1977, Figure 7/16 (p. 116)):

The radiological effects of an extremely improbable nuclear meltdown accident was determined for one German location using the accident consequence model (status 1976) selected for the German Reactor Safety Study. This is characterized by release category PWR 2 corresponding to WASH-1400.

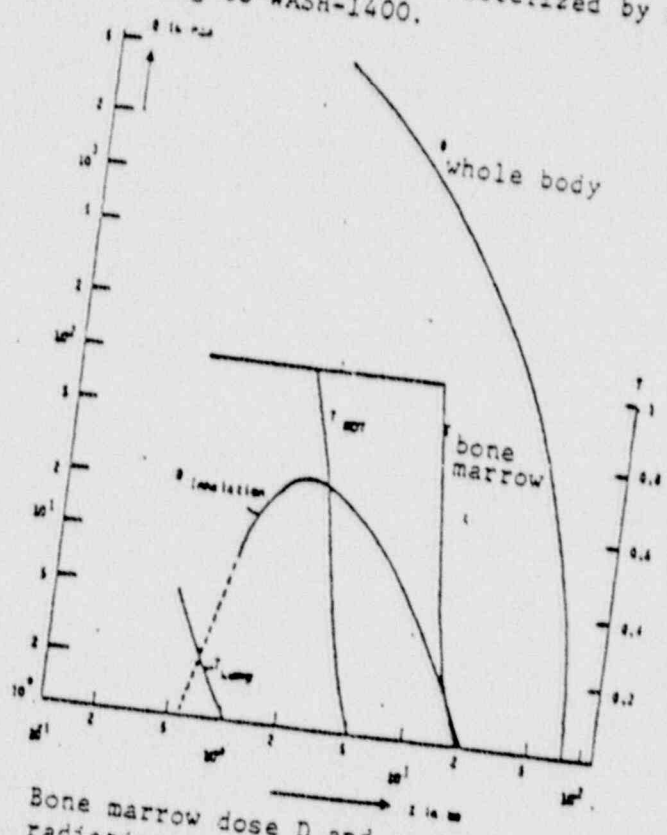


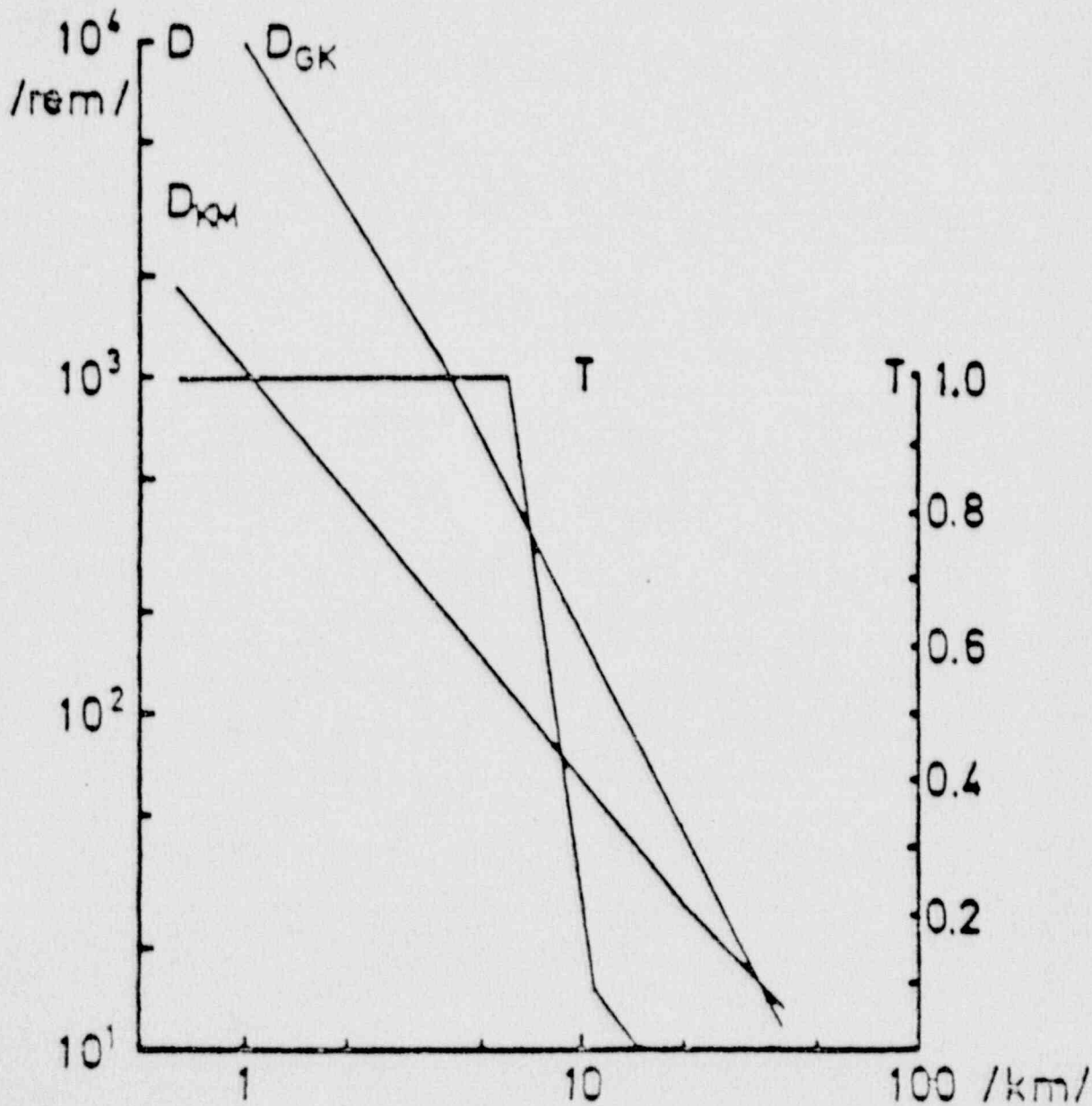
Figure 7/15. Bone marrow dose D and probability T for acute radiation death with a 24-hour dwell.

The result is published in Figure 7/15 of the above annual report. From this, there results a limiting radius for mortality of 12 to 20 km in the case of average weather conditions (category D) with rain.

Our own estimates were carried out for the PWR 2 nuclear meltdown accident for a near-to-ground release without consideration of thermal lift, most unfavorable propagation conditions (weather condition T, wind velocity 1 m/s) and constant wind direction during the entire emission time. In order to make the a concerning the range with early mortality analogous to the German and American risk

studies, for the abovementioned limiting conditions, the bone marrow dose owing to inhalation was computed for an integration time of 30 days D_{BM} . In addition, the radiation exposure of the whole body owing to external irradiation from fission products deposited on the ground (deposition rate 10^{-3} m/sec for aerosols, 10^{-2} m/sec for iodine) was computed for a dwell time of 24 hours D_{GK} and the resulting summation curve was prepared.

The results are depicted in the following figure:



The limiting radius for early mortality accordingly lies in the area between ca. 8 and 15 km. Herewith, none of the effects reducing the consequences are considered such as, for example:

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Stay in house or in dwelling or in the open (shielding of buildings),

Emergency protective measures (for example, computation of external irradiation over 24 hours),

Thermal lift and the thereby-conditioned reduction of the concentration near the ground.

In addition, the whole-body dose and the bone marrow dose owing to inhalation was computed for the abovementioned limiting conditions for an integration time of 50 years in order to explain the discrepancies between the computed radiation exposures of Work Report 290 and the results mentioned in technical literature. The values computed herewith produce radiation exposures for a distance of 10 km which a human being keeps for a period of 50 years after a one-time absorption of fission products through inhalation.

	AB 290	New Computation
Release	Release from core + release from plant	Release factors according to WASH-1400
Propagation parameters	Pasquill	Vogt
Dose contents	Original data	Data according to US NRC Regulatory Guide 1.109
Whole-body dose/ rem/integration time 50 years	$9.4 \cdot 10^4$	$8.4 \cdot 10^2$
Bone marrow dose/ rem/integration time 50 years	$1.8 \cdot 10^6$	$3.3 \cdot 10^3$

The differences in the doses determined in the old Work Report 290 and in the newer computations are to be attributed to the abovementioned differences in the release, propagation parameters and dose constants.

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The release factors according to WASH-1400 take into consideration the deposition effects always present during transport of nuclides from exit from the fuel through the containment until passing into the outside environment. This effect was not taken into consideration in Work Report 290 because, without an accurate detailed knowledge of the nuclear decommissioning center, no corresponding factors would have been derivable for the nuclear decommissioning center.

The propagation parameters of Vogt better reproduce in accordance with the most recent knowledge the propagation conditions in the Federal Republic of Germany (rather considerable rough terrain). They are a constituent of the "general computational bases for determination of radiation exposure owing to emission of radioactive substances with exhaust air" which was to be used by the Federal Ministry of the Interior as a basis for a legal ordinance to Section 45 of the Radiation Protection Ordinance. They are also being used at the present time in certification practice.

The dose constants are derived from the currently valid NRC Regulatory Guide 1.109. Dose constants from the newest model computations as they are used, for example, in WASH-1400 would reduce the above dose values.

The new computations show, for example, that the results published in Work Report 290 -- which at that time were only to be used for purposes of comparison -- cannot be used for absolute statements with respect to radiological danger even on an approximative basis. At this point, reference should again be made to the conservative limiting conditions used for the new computations. The computed doses are not suitable for making statements concerning the risk of nuclear energy plants.

EXPLANATION OF ABBREVIATIONS

NEZ Nuclear Decommissioning Center
BE Fuel element
HAW High level radioactive waste
DWR Pressurized water reactor

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