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Abstract

In order to quantify the nuclide specific emissions from the secondary loop of light-water reactors during normal operation and as a result of selected incidents the computer program SEKEM4 was developed.

The program is based upon a transport model, which describes the nuclide specific activity transport of fission and activation products from the primary loop through the secondary loop to the environment (air pathway). The program was tested with the aid of a sensitivity analysis, and the effect of parameter changes on emission rates was compared.

In connection with the analysis of the transport phenomenon involved it was observed that not only the residual humidity transport but also the nuclide transport in the steam phase must be considered. The development of a detailed phase distribution model, which integrates commonly used formulas, has taken this factor into account.

Within the frame of the secondary system model a previously unnoticed transport path was considered and its impact on the emission rate of the secondary loop was determined. The blowdown bound steam generator water must pass the blowdown flash tank. The steam generated thereby is diverted around the blowdown filter directly to the feedwater tank. Recognition of this pathway results in an increase in the emission rate by a minimum factor of 2.

Within the framework of the analysis of the primary loop concentrations of fission and activation products it was established that the number of relevant nuclides is much larger than previously assumed, and that especially essential trace elements must be carefully accounted for in emission and immission calculations.

According to our research as well as our own calculations we conclude that the importance of normal operation leakages as a result of defective steam generator heating tubes is being underestimated.

Even steam generator leakages of 25 to 50 kg/hr can lead to main steam concentrations of I-131 which exceed the official limits. As a result the allowable marginal immission values for infants can be exceeded after only 20 hrs by a factor up to 120.

As the shutdown value of the N-16 signal of the secondary loop is not dependent upon the reactor power a reduced functioning of the signal with respect to the reactor safety system must be assumed.

Of the quantitatively analyzed accidents the incident: steam generator heating tube rupture (10 heating tubes) with relief valve failure in open or semi-open position was studied under different conditions. Applying the single fault criterion this accident must be considered to be a design basis accident. The calculations show that this accident also results in exceeding the marginal values for adults by a factor of 60.

Table of Contents

1. Introduction

The purpose of the present study is to develop a computer program for determining the transport of activity in the secondary loop and the release of activity through the power house. Secondary loop emissions have not yet been recorded by measurement instruments and are due to steam generator (SG) heating tube leaks. In Germany SG leaks occurred as early as in 1971 in the Obrigheim NPP, and while this study was underway leaks occurred in the Biblis and Stade NPPs. In the Ginna reactor (U.S.A.) there were such massive SG leaks in February of 1982 that the secondary-side relief valves were activated and activity was released into the environment.

The program is tested with the aid of a sensitivity analysis, and the effect of the following parameters on the emission rates is compared:

> Primary loop activity, SG heating tube leaks, decay constant, steam emission rates from the high-pressure section and from the feedwater degasification system, residual moisture (for instance, in the SG), blowdown rate per SG, decontamination factors in the blowdown demineralization system and phase distribution factors in the high-pressure and low-pressure sections.

In addition, load changes and activity spikes are studied. In the sensitivity analysis, as well, the effects of different parameters on emission pathways which have not yet been considered is studied.

It is planned that realistic calculations will be carried out for operational leaks in the SG. In this process particular attention is to be paid to studying the extent to which the officially determined maximum I-131 main steam concentrations can be observed and, if necessary, what the effect of violation of these values is. For this purpose the emissions are to be evaluated by means of immission calculations. A study will be made in particular of whether other nuclides aside from I-131 are relevant in secondary loop emissions.

The operability of the N-16 signal, and in particular the relationship between the response value and reactor power are being studied.

The single-fault concept is applied to loss of coolant accidents [LOCA] due to SG heating tube failure. A study is made of whether the resulting accident sequences are contained in already-discussed accidents and what the consequences can be. The emissions of these accidents are calculated with the aid of the computer program.

2. System Analysis

- Introduction

Extensive literature studies show that the secondary systems of German NPPs with PWRs are comparable in terms of their essential components. Compare [1], page 61, Figure 3.2.8, Basic Circuit Diagram of the Primary Heat Removal System, [2], Figure 8, General Diagram, [3], Figure 6, Vapor Circuit, [4], in particular page 12, [5], in particular page 32, Figure 1. Two-loop, three-loop or four-loop plants are, in principle, constructed identically so that in emission calculations through the powerhouse general only the different mass fluxes and the different outputs of the blowdown demineralization systems need to be taken into account, compare [6], [7], [8], [9], [10].

In the study of individual components, among other things, the standard work of Karl Schroeder [11] was taken as the point of departure.

In the model development, direct contact with operators was essential, along with the literature studied. For example, let us mention here the inspection of in-house documents at the Neckarwestheim Atomic Power Plant (AKW) as well as on-site inspections and discussions at the Grafenrheinfeld and Biblis A and B NPPs.

The reference plant selected was the Biblis NPP, Unit B (compare [12], page 35) although many of the components of power plants built prior to that time are not designed with the corresponding safety. Unit B has the advantage, however, that it largely corresponds to plants presently under construction or in planning (line 8) so that the computer program developed by us can also be applied to them.

Below we will first make a detailed examination of the study's model of the secondary loop of lightwater reactors [LWRs]. In this process the system-induced and work-related boundary conditions and the resulting simplifications are discussed. Subsequently, the values which determine the nuclide transport in the secondary loop (namely the input parameters which are decisive for the computer program) are considered.

For the nuclide distribution factors, a mathematical form for site-specific and nuclide-specific distribution factors is first derived separately. At the end of this chapter there is a section on the difficult process of information procurement.

2.1. Model of the Secondary Loop of LWRs

To calculate and simulate radioactive emissions from the secondary loop of a PWR, it is necessary to develop a model of the secondary loop. The model must fulfill two conditions:

1. On the one hand, it must reflect the basic structure and function of the secondary loop in a simplifying and summary way without, at the same time, omitting or distorting significant features of the system in their functional relationship.

2. The model must make it possible to mathematically formalize the functional system relationships.

The secondary loop model is thus the result of a systems analysis of the seondary loop. It must contain the essential transport pathways for radionuclides and, in particular, emission pathways.

2.1.1. Structure and Function of a PWR

The description of the secondary loop (also called the feedwater steam loop) will be preceded, for the sake of completeness, by an explanation of the structure and operation of a NPP with a PWR.

The structure and operation of a NPP with a PWR are illustrated by the simplified operating diagram in Figure 2.1.1. Here we see three loops:

-- The Primary Loop

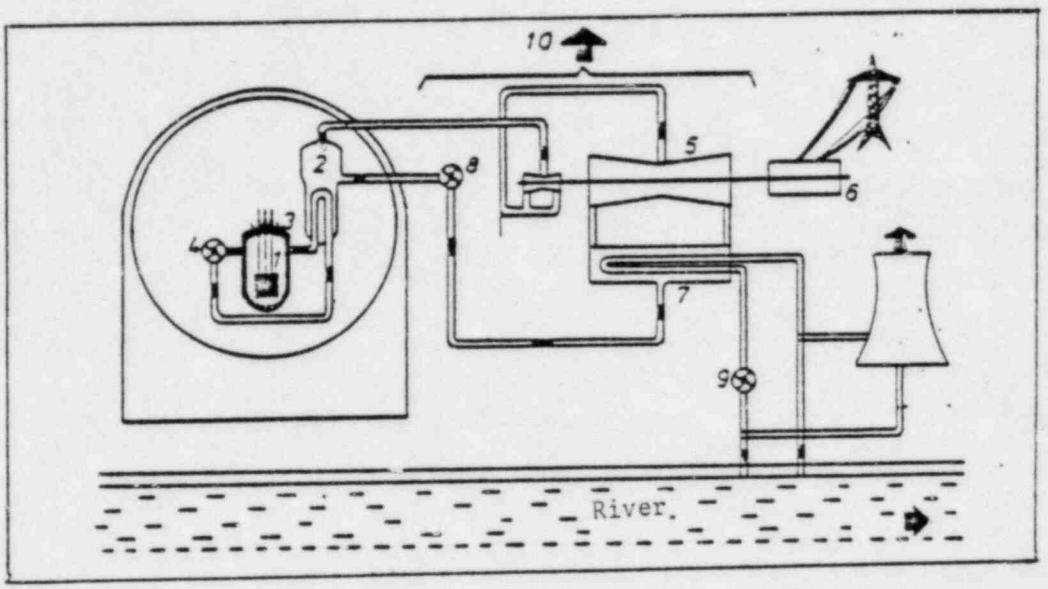
The essentially self-contained reactor coolant loop in which reactor coolant pump 4 transports the heat generated in reactor core 1 from the reactor pressure vessel [RPV] 3 through SG 2 (mostly between 2 and 4 SGs per reactor unit) to the secondary loop.

-- The Secondary Loop

The feedwater cooling loop corresponds largely to that in conventional thermal power plants. In it the main feedwater pump 8 pumps the feedwater into the SG where it is evaporated by absorbing heat from the reactor loop. The steam which develops drives turbine plant 5, which consists of one high-pressure and three parallel-connected low-pressure turbines, and the turbine plant drives generator 6. The steam emerging from the turbine plant is liquefied in condenser 7 by releasing its heat to the main cooling system, and the loop is closed. Arrow 10 indicates that steam leakages through the power house roof continuously occur during operation.

-- The Tertiary Loop

The main cooling water loop serves the purpose of removing heat from the evaporator overhead condenser. In this process the coolant pumped through the evaporator overhead condenser by main cooling water pump 9 absorbs approximately two-thirds of the heat generated in the reactor core and releases it either directly to a river or the sea or, via cooling towers, to the atmosphere, depending on the ambient conditions.



1 Reactor core
2 SG
3 RPV
4 Reactor coolant pump
5 Turbine
6 Generator
7 Evaporator overhead condenser
8 Main feedwater pump
9 Main cooling water pump
10 Steam emissions from the power house

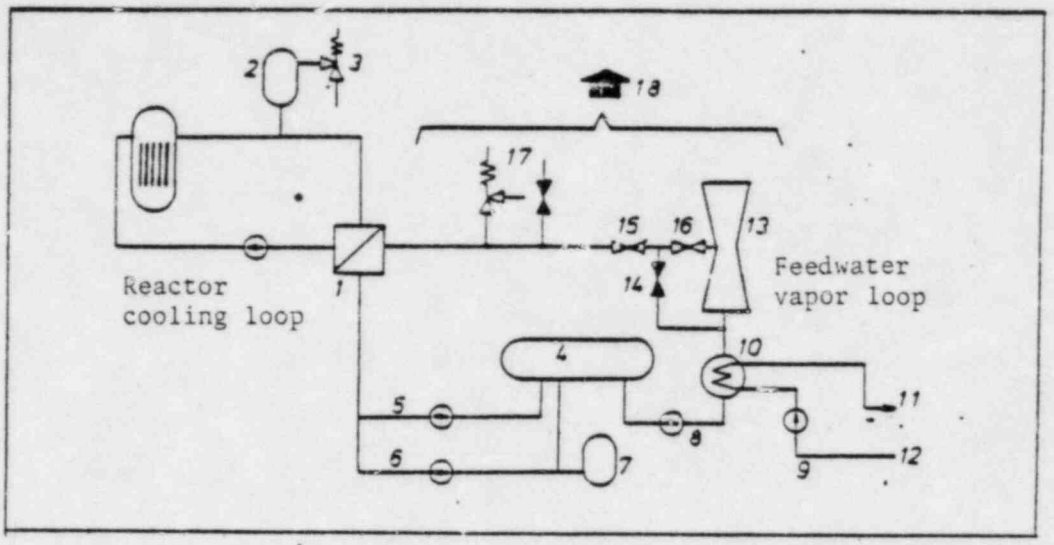
Figure 2.1.1. Functional Diagram of an NPP with a PWR.

The generation of electrical current from heat is done in NPPs in basically the same way as in other thermal power plants. The difference involves the generation of heat by the process of nuclear fission. Since in this process several hundred radioactive isotopes are generated and these isotopes represent a health risk because of their radioactive radiation and the associated possible radiation damage to biological organisms, special precautions must be taken to minimize the release of radioactive substances to the environment of the NPP. It is impossible to completely prevent radioactive emissions even with undisrupted normal operation.

2.1.2. Structure and Function of the Secondary Loop

The structure and function of the secondary loop are presented in simplified form in basic circuit diagram 2.1.2.

The function of the feedwater-steam loop consists of the removal of nuclear heat and the thermokinetic conversion of it into electrical current. The decisive functional elements in this case are the SGs and turbine set with generator and evaporator overhead condenser.



1 SG
2 Pressurizer
3 Pressurizer safety valves and relief valves
4 Feedwater tank
5 Main feedwater
6 Emergency feedwater system
7 Demineralized water tank
8 Main condensate
9 Main cooling system
10 Evaporator overhead condenser
11 Sealing pit
12 Inlet structure
13 Turbine
14 Main steam bypass
15 Main steam slide valve
16 Turbine control valve and quick-action stop valves
17 Main steam safety valves and blowdown control valves
18 Steam emissions from the power house

Figure 2.1.2. Basic Circuit of the Reactor Cooling Loop and the Feedwater-Steam Loop.

As already briefly outlined, the main steam generated in the SGs leaves the containment via the main steam pipe which is equipped with relief and safety valves and drives the turbine set through turbine valves as a function of output. The subsequently depressurized steam is condensed in the evaporator overhead condensers, the condensate is collected in the condensate tank and recycled by main condensate pumps through several preheating stages into the feedwater tank and from there, by the main feedwater pumps, back into the steam generators. In this process, as is usually done in thermal power plants, steam is diverted at various turbine stages to preheat the condensate or feedwater.

A complete technical description of the secondary loop is given within the framework of the circuit diagrams for:

1. Reactor coolant loop
2. Main feedwater system
3. Emergency feedwater system
4. Demineralized water system
5. Main steam pipe system

Each of these circuit diagrams comprises approximately 6-8 DIN A 4 pages (compare German Reactor Safety Study, [15]) with several hundred components.

At this point it is not reasonable either in terms of content or work to go into the amount of detail even in these circuit diagrams for the determination of secondary loop emissions. As the following section shows, it is possible to reduce the host of components to an adequately small number of components for the purposes of model consideration.

2.1.3. Development of the Working Model

The following general goal corresponds to the problem stated for the present study:

-- With a preset primary loop activity of an arbitrary radionuclide as well as SG leakage, the specific activity of this nuclide, the activity flows between connected points and the resulting emissions from the secondary loop at any time and at any place are to be calculated.

Within the framework of the systems analysis, the following restricted goal function is decisive for the development of the model:

-- With a preset primary loop activity of an arbitrary radionuclide as well as SG leakage, in the equilibrium case and for certain operating transients the specific activity of this nuclide at points which are relevant for the emissions and the resulting emissions from the secondary loop are to be calculated.

The task of the development of the model thus consists of providing a model which is capable of stimulating, with adequate precision, the continuous space/time distribution of an arbitrary radionuclide in the secondary loop. In this process, as regards the time distribution both the steady-state activities and, in special cases, the time-dependent transient conditions are to be considered. With regard to distribution in space, an appropriate selection is to be made of the points in the feedwater steam system.

It must also be determined what values are essential to nuclide transport in the secondary loop.

Within the framework of the present model development, the following nuclide-specific and non-nuclide-specific specification factors are used as a point of departure:

1. Specific primary loop activity and SG leakage

The specific primary loop activity is nuclide-specific, while the SG leakage is not. These two values are linearly incorporated into the activity calculations. They represent boundary and initial conditions which directly affect the level, but not the space/time distribution of the specific nuclide activity.

2. Mass flows and steam emissions

Mass flows and steam emissions are not nuclide-specific. The radionuclides carried through the SG leak into the secondary loop are water-borne or

steam-borne; they are thus distributed in accordance with the mass flows of the water and steam phases. This is also reflected, among other things, by the fact that for each radionuclide its specific activity, i.e., the number of decays per unit of time and unit of mass (usually in Ci/t) is calculated at every point in the secondary loop which is regarded as relevant.

3. Phase distribution factors α or decontamination factors DF

The α and DF factors are nuclide-specific and indicate in what proportions a radionuclide remains in the liquid or gaseous phase in the event of phase transitions of evaporation or condensation. Thus, for example, a nuclide which is present in ion form will always remain for the most part in the liquid phase in an evaporation process. The α and DF factors depend on the chemical and physical properties of the nuclide under consideration as well as on the technically pre-established values pressure, temperature, air humidity, pH value, etc.

A detailed description of the values which determine nuclide transport is given in Sections 2.2 and 2.5 of the systems analysis.

Since the model development is closely associated with mathematical formalization or the mathematical description and solution postulates, let us skip ahead here to Section 3, Computer Program. The following basic physical assumptions determine the mathematics used:

1. The principle of conservation of mass or the continuity equation for mass flow or mass flow density is applicable.

2. In the secondary loop system (model) there are no sources of radionuclides other than the SG leaks, determined by the leakage rate in the SG and the nuclide-specific primary loop concentration.

3. For every point in the secondary loop it is possible to compile an activity flow balance from the technically pre-established mass flows in the form of a linear differential equation. The activity distribution for all points is found as the solution to the differential equation system (DE system).

Accordingly, the number of points to be considered directly determines the scope of the DE system and the expense involved in solving this system.

The following points are selected with the above in mind and are briefly explained (compare Figures 2.1.3 and 2.1.4).

1. SG leaky

2. SG intact

3. Main steam

4. High-pressure turbine

5. Water separator/intermediate superheater

6. Low-pressure turbines

7. Evaporator overhead condenser

8. Hot well

9. Feedwater tank

10. Blowdown flash tank

11. Blowdown demineralization system

12. Sump

13. Atmosphere

-- Re 1 and 2: The SGs play a central role tn two ways: on the one hand, the SGs are the point at which the radioactivity is transferred from the primary loop into the secondary loop. On the other hand, depending on the element and the chemical form in which it comes, there can be a build-up of the activity in the liquid phase in the evaporation process. In this process, pressure, temperature and residual moisture are of decisive importance.

This phenomenon, which is due to the fact that the solubility of the substance is different in water than in steam, is described by the phase distribution and decontamination factors mentioned above (compare Sections 2.2 and 2.5).

It is to be assumed that a SG leak will not occur in all SGs simultaneously. Thus, in practice, SG leaks have occurred to date in only one, or at most in two of the usual four SGs. The model must therefore provide for precisely two "SG points", of which the one represents the intact SGs and the other the SGs with heating tube leaks.

-- Re 3: The main steam which is generated in the SG and is largely dried is distributed in an extensive pipe system (compare Figure 2.1.3). The soundness of introducing a "main steam" point is due to the consideration that the main steam pipes of the two SG points combine into a collecting pipe prior to the inlet into the turbine and thus affect the activity distribution. In addition, a portion of the already collected main steam is branched off and is sent, for purposes of preheating, past the turbine to the fifth point, the water separator and intermediate superheater, and from there to the feedwater tank. Because of the feedwater degasification device, this is important for the emissions calculation.

-- Re 4: The high-pressure turbine is to be regarded as a relevant point to the extent that, on the one hand, due to its numerous taps which are combined by us it represents a mass flow distribution node with direct influx into the feedwater tank and since, on the other hand, values which are decisive for the phase distribution factors such as pressure, temperature and residual moisture or steam content, vary at this point.

-- Re 5: As in the case of the high-pressure turbine point, the water separator/intermediate superheater represents a mass flow branching with direct influx to the feedwater tank which affects the activity flows by means of the α factor. By the same token, the pressure, temperature and moisture content of the main steam are influenced by these components.

-- Re 6: See 4.

-- Re 7: The evaporator overhead condenser is a power plant component which has an operational and a safety task since it plays a decisive role in heat removal, for instance in the event of a reactor scram and the associated turbine trip and operation through a bypass station. The condenser also represents the interface with the tertiary loop, the main water loop. Evaporator overhead condenser leaks decisively affect the secondary loop water chemistry and consequently the SG heating tube corrosion damage.

Finally, a significant emission pathway for noble gases from the secondary loop runs through the evaporator overhead condenser exhaust.

-- Re 8: The hot well represents a mass flow node. Here the mass flows from the blowdown demineralization, low-pressure turbine taps and condensate collect and flow from the hot well into the feedwater tank. The demineralized water for making up for the water and steam losses is essentially fed into the hot well.

-- Re 9: There is a direct emission pathway from the feedwater tank to the atmosphere through the feedwater degasification system. The feedwater tank is also a mass flow node, with a direct line to the SGs.

-- Re 10: The blowdown flash tank serves to adapt the pressure and temperature of the SG water to the values required for the blowdown system. In the depressurization of the SG water, steam is generated which is fed as heating steam directly to the feedwater tank. This mass flow is decisive for the activity in the feedwater tank and for its emission.

-- Re 11: The blowdown demineralization system serves to retain the radionuclides which are present primarily in the form of salts.

-- Re 12 and 13: The points sump and atmosphere represent the aqueous or vaporous sinks.

The above-described model optimization takes into account in summary form the following boundary conditions:

-- Degree of detail of the description of the reactor equipment,

-- Specification factors for readionuclide transport,

-- Mathematical formulation approach,

-- EDP optimization which is appropriate to the problem (storage space, computer time, etc.).

2.1.4. The Secondary Loop Model for Calculating Emissions of Radioactive Nuclides

The results of the development of the model are presented in Figure 2.1.3. As the mass flow balances show, the components and pipe systems which are relevant for the mass flows to be considered are encompassed therein. The model of the secondary loop for the computer program is derived from Figure 2.1.3 and presented in Figure 2.1.4.

A comparison of Figures 2.1.3 and 2.1.4 shows that control and safety components such as control valves, isolation valves, safety valves, parallel piping are

either omitted in Figure 2.1.4 (since they are of no importance for nuclide transport) or are combined.

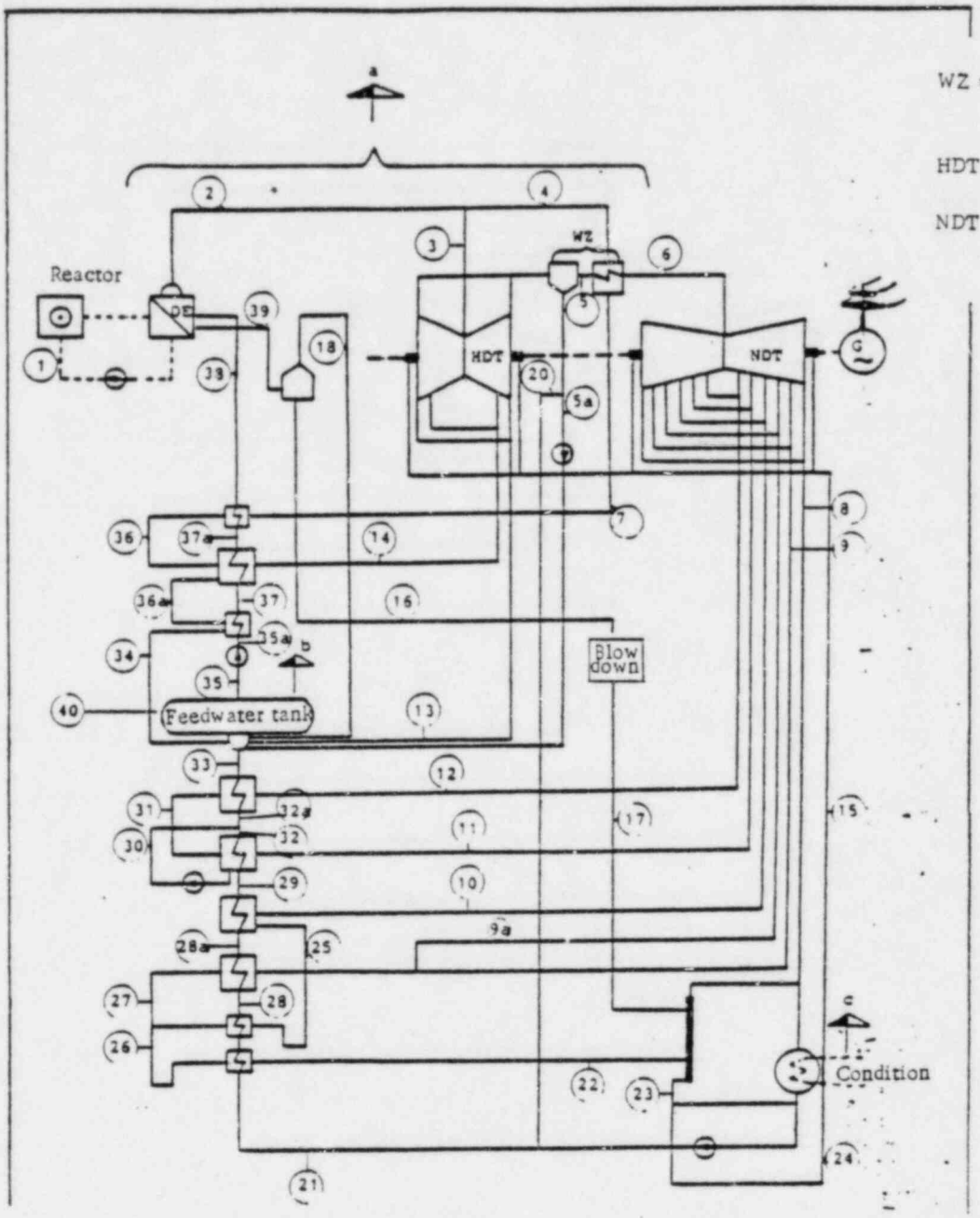
Three release pathways into the atmosphere, two of which do not run through the vent stack, are of particular importance for the emissions:

1. The main leak site is the high-pressure section from which emissions to the ambient air constantly occur, even during normal operation. In Figure 2.1.4 these leaks are combined by a set sign (——,——). In particular, this is construed to include all emissions from the following components: high-pressure drain/tank with standpipe to the power house roof; leaks from different seals and stuffing boxes.

2. Emissions from the feedwater degasification system. Due to considerations of water chemistry, the feedwater is cleared of dissolved corrosive gases such as oxygen and carbon dioxide as much as possible. The gases are removed directly through the power house roof with steam as the carrier gas (without a vent cooler/vapor condenser).

The important point here is the fact that only approximately one-half of the mass flow, which is tapped off from the SGs to the flash tank of the blowdown demineralization system for purposes of blowdown, reaches the ion exchangers at all. The other half of this mass flow reaches the feedwater tank unfiltered. In the event of SG leaks, activity reaches the environment directly via this pathway through the feedwater degasification system (compare Figure 2.1.5).

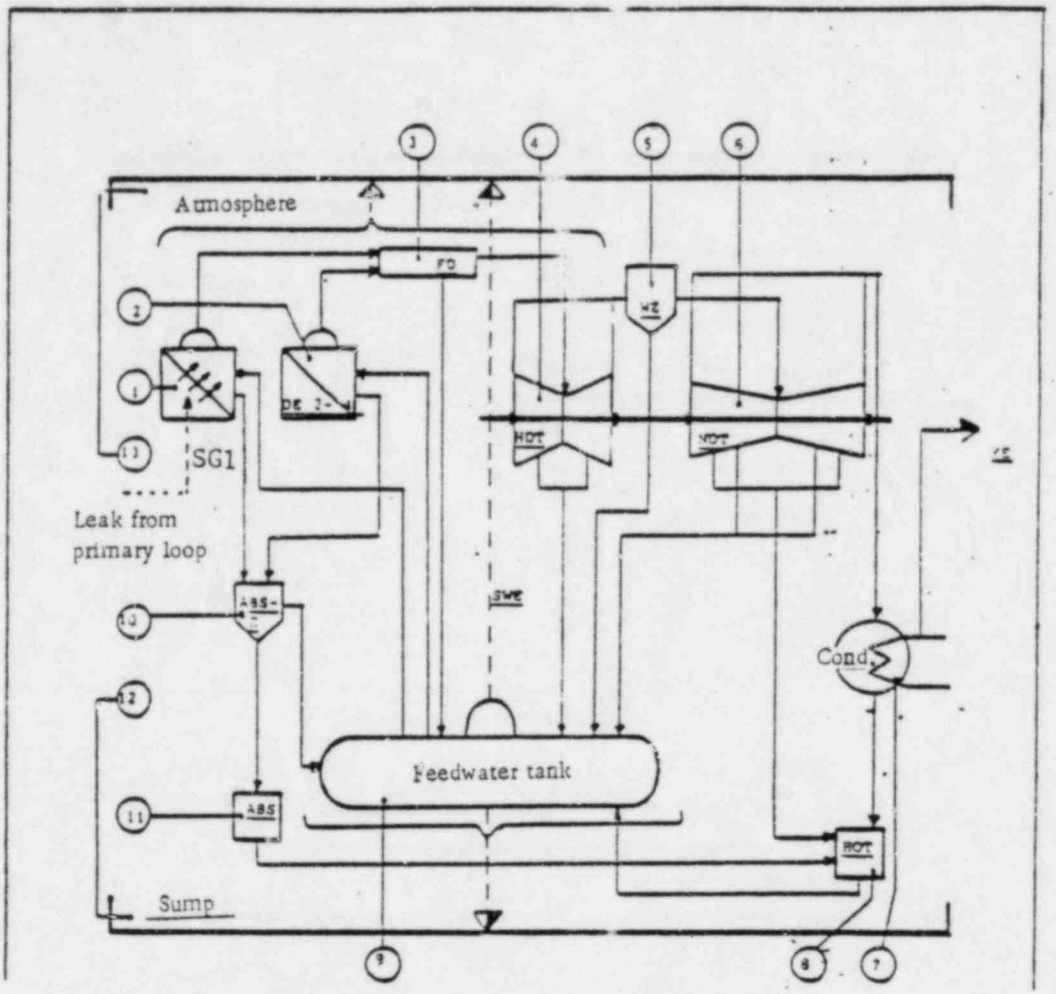
3. Emissions from the evaporator overhead condenser exhaust. In the process of maintaining an underpressure, the noble gases are sent on to be emitted through the vent stack.



a - Emissions in the high-pressure section
b - Emissions in the feedwater degasification system
c - Emissions in evaporator overhead condenser evacuation through the stack

} Unfiltered

Figure 2.1.3. Simplified Model for Calculating Secondary Loop Emissions in the Event of SG Leaks.



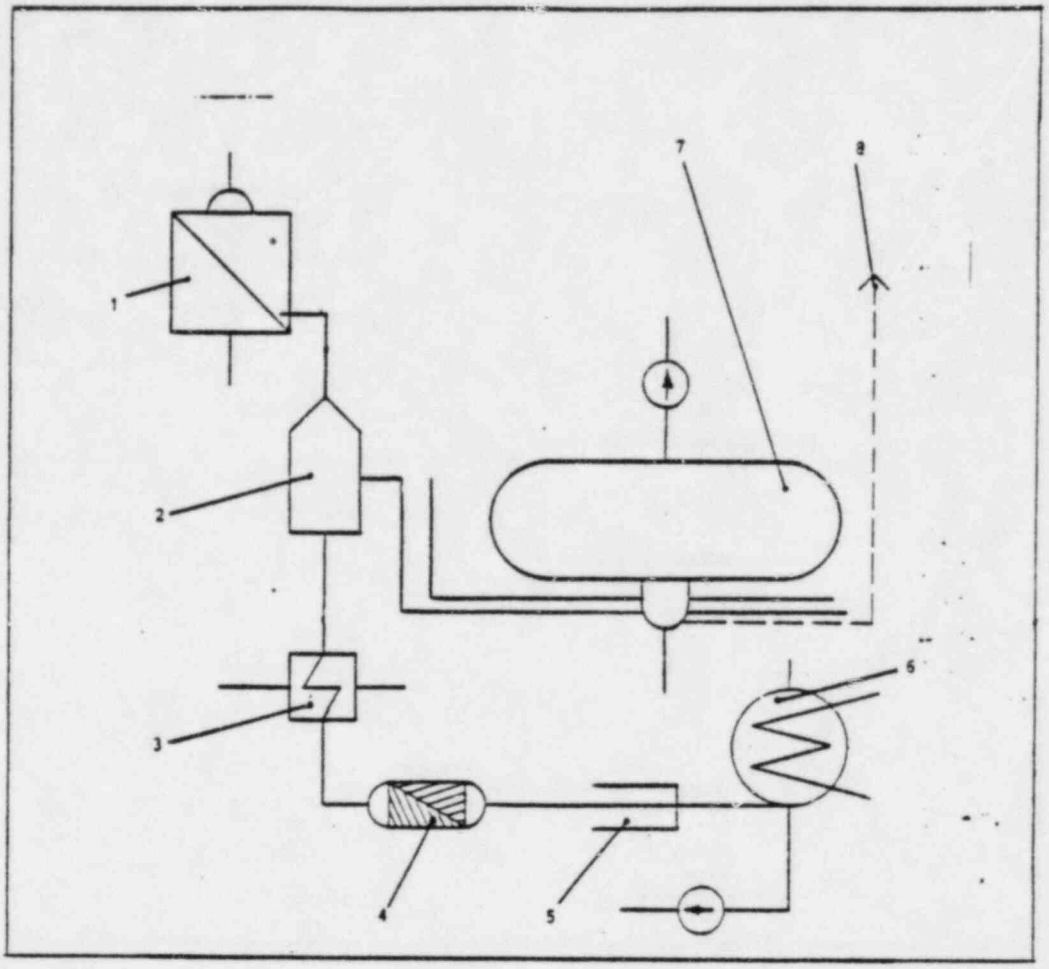
NDT = Low-pressure section
HDT = High-pressure section
FD = Main steam
W2 = Water separator-intermediate superheater
ABS = Blowdown
SWE = Feedwater degasification
ABS-E = Feedwater flash tank

Points in the computer program

1 SG leaky
2 SG intact
3 Main steam
4 High-pressure turbine
5 Water separator-intermediate superheater
6 Low pressure turbine
7 Evaporator overhead condenser
8 Hot well
9 Feedwater tank
10 Blowdown flash tank
11 Blowdown demineralization
12 Sump
13 Atmosphere

SWE = Feedwater degasification
KE = Condenser evacuation

Figure 2.1.4. Model of the Secondary Loop for the Computer Program.



1. SG
2. Flash tank of blowdown demineralization system
3. Blowdown demineralization cooler
4. Purification system of blowdown demineralization
5. Hot well
6. Evaporator overhead condenser
7. Feedwater tank
8. Emission pathway: feedwater degasification → atmosphere

Figure 2.1.5. SG Blowdown Demineralization - Diagram and Mass Flows.

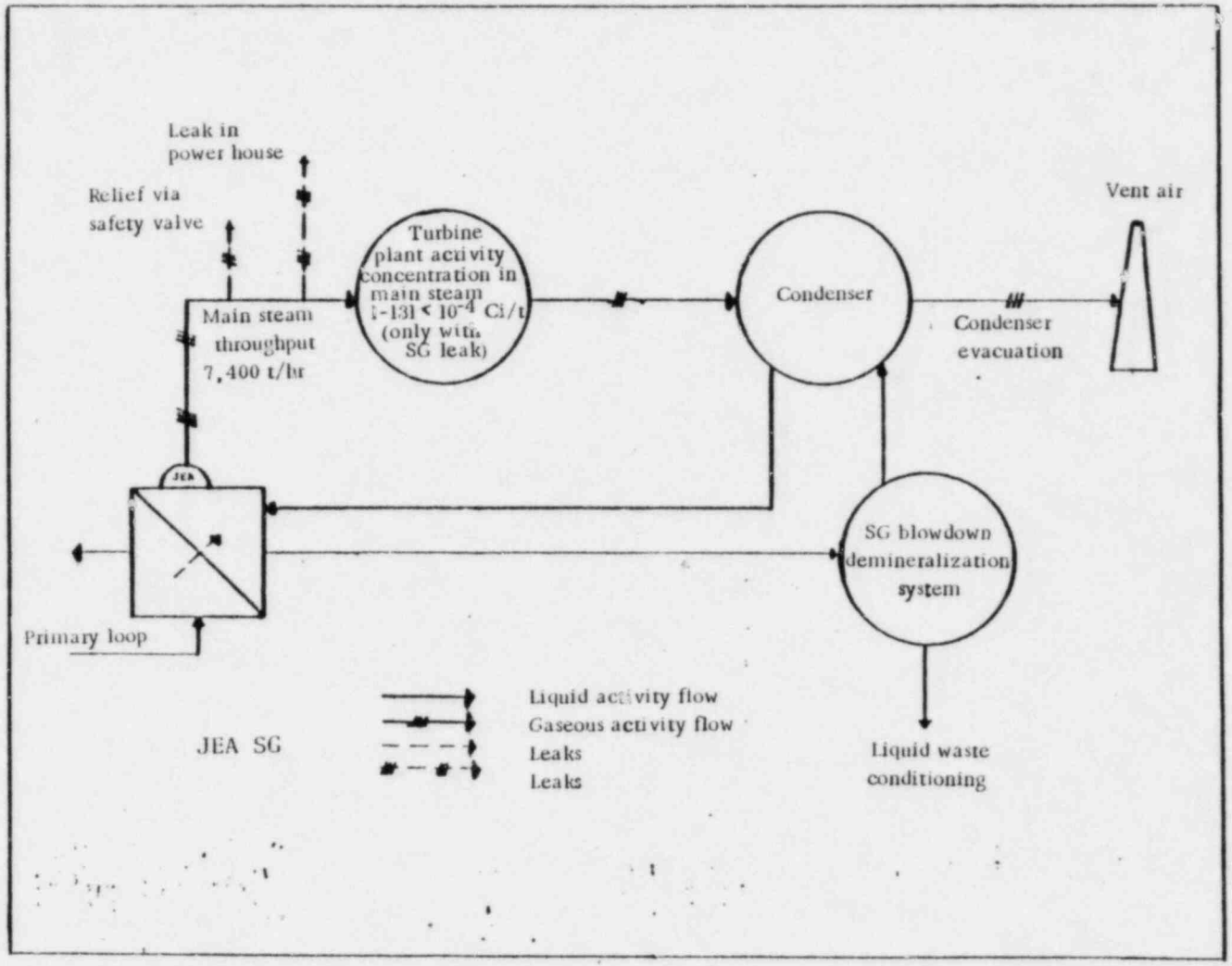


Figure 2.1.6. Illustrative Diagram of Activities in Systems (According to [13]).

2.1.5. Systems-Theory Comparison of the Secondary Loop Model with the Secondary Loop Model of the KWU

To conclude the systems analysis, we will now briefly discuss the secondary loop model used by the KWU [13]. The systems-theory comparison is informative in many respects: on the one hand, because the KWU model has been incorporated in different ways into the evaluation of nuclear power plants within the framework of the licensing procedure under the Atomic Law. On the other hand, the comparison sheds light on the importance of the systems analysis since this makes it clear that in principle a model cannot take into account more emissions pathways than were previously considered within the framework of the systems analysis. This means that without completely considering all relevant system values (here nuclide transport including the emissions pathways) the derived model will also be unable to completely consider the emissions in question.

The KWU diagram for calculating activity flows in the secondary loop is presented in Figure 2.1.6.

The comparison shows in particular:

1. The KWU model does not contain a feedwater tank, but rather only the evaporator overhead condenser, and thus no feedwater degasification. Consequently, all of the main steam runs through the turbines to the condenser, and this necessarily leads to an overestimation of the emissions which are released through the stack, i.e., under "monitored" conditions.

2. The KWU model does not provide any blowdown demineralization flash tank and thus eliminates a significant contribution of activity to the direct unfiltered emission through the power house roof.

3. The only direct emission pathway which the KWU model provides for is relief through the safety valves. It does not consider any emissions from the high-pressure section of the secondary loop. These two factors lead to a significant underestimation of the direct and unfiltered emissions to the outside during normal operation (no relief).

4. The diagram suggests that the leaks enter the power house (with the exception of relief through the safety valve). In contrast, however, the text description does indicate I-131 emissions through the power house roof (and through the liquid waste), but this is based on the completely false assumption that the N-16 scram system will limit the I-131 concentration in the main steam to the values established by the Federal Minister of the Interior. This point is discussed in greater detail under the section on "SG Leaks".

In summary, it can be stated that the KWU model necessarily leads to a large underestimation of the emissions.

2.1.6. Bibliography for 2.1

1. Smidt, D., "Reaktor-Sicherheitstechnik" [Reactor Safety Engineering], Berlin, Heidelberg, New York, 1979.

2. KWU (publisher), "Das Kernkraftwerk Biblis des RWE, 10, Musteranlagen der Energiewirtschaft" [The Biblis NPP of the RWE, 10, Pattern Systems of Energy Management], Muelheim, after 1976.

3. Bald, A. and O. Brix, "The Steam Power Plant of the Biblis NPP", ATOMWIRTSCHAFT, ATOMTECHNIK, Vol. 19, (8/9), pp. 431-438, 1974.

4. KTA: "Messung fluessiger radioaktiver Stoffe zur Ueberwachung der radioaktiven Ableitungen. Sicherheitstechnische Regel des KTA", [Measurement of Liquid Radioactive Substances to Monitor Radioactive Releases. Safety Engineering Rule of the KTA], KTA 1504, Version 6/78.

5. VGB (publisher), VGB NPP Seminar, 1970. Papers. General Theoretical Principles. PWRs. Primarily type-dependent technology in the NPP. Essen, 1970.

6. ATOMWIRTSCHAFT-ATOMTECHNIK, Vol. 12, December 1968: "The Obrigheim NPP, From Planning to Start-Up", Duesseldorf, 1968.

7. ATOMWIRTSCHAFT-ATOMTECHNIK, Vol. 11, November 1971: "The Stade NPP", Duesseldorf, 1971.

8. Sparhuber, R., "The Reactor and Steam Power Plant of the 660 MW Stade Major NPP", ENERGIE, Vol. 20, No. 4, pp. 100-107, 1968.

9. ATOMWIRTSCHAFT-ATOMTECHNIK, Vol. 8/9, Aug./Sept., 1974: "The Biblis NPP", Duesseldorf, 1974.

10. Schroeder, H. J., A. Dorr and G. Paffrath, "Chemistry in the Biblis NPP - Operating Experience", VGB KRAFTWERKSTECHNIK, Vol. 60, February 1980.

11. Schroeder, K., "Grosse Dampfkraftwerke, Ausfuehrung und Bau" [Large Steam Power Plants, Design and Construction], Vol. 3: Power Plant Outfitting, Parts A and B, Heidelberg/New York, 1966 and 1968.

12. "Deutsche Risikostudie Kernkraftwerke" [German Risk Study for NPPs], A Study of the Risk Caused by Accidents in NPPs, Cologne, 1979.

13. Stuenkel, H., G. Hecht and E. Doerr, "Technical Conference on Radioecology, Retention of Radionuclides", German Atomic Forum (KWU), Bonn, October 2-3, 1979.

14. KWU (publisher), "Kernkraftwerk mit 1300-MWe-Druckwasser-reaktor (Standard-Anlage)" [NPP with 1300 MWe PWR (Standard Plant)], Illustrative Diagram of Activities in Systems, Company secret, 1977.

15. "Deutsche Risikostudie Kernkraftwerke" [German Risk Study on NPPs], Technical Vol. 2/II: Reliability Analysis, Cologne, 1980.

2.2. Mathematical Model of the Phase Distribution and Decontamination Factors

2.2.1. General Consideration

The nuclide distribution in the secondary circuit is determined by the mass flows of the transport medium water. The transport medium water is present in the secondary circuit either predominantly in the liquid or predominantly in the gaseous phase. The solubility of the nuclide or the chemical compound in which the nuclide occurs in water or vapor is accordingly the value which primarily determines nuclide transport and thus nuclide distribution.

Solubility data are always data on the ratio of the quantities of a substance or element which are dissolved in the solvent (here water or steam) under constant conditions to the quantities of the same element or substance which remain insoluble under these conditions. Solubilities are thus statements of concentration ratios.

The solubility data shown in many standard works ([1], [2]) apply to a static equilibrium in which the observation time is thus long compared to the establishment time of the concentration equilibrium.

Because of the high [large] mass flows in the secondary circuit, it can be assumed that only dynamic equilibria are present. These concentration conditions are constant over time, as in the case of static equilibrium, but, in addition to the chemical conditions and the values of pressure and temperature, they are also dependent on physiotechnical conditions such as the dynamic boundary layer thickness of the interface, etc. This system of specification factors which generates the dynamic solubility equilibrium itself varies within the secondary circuit so that, to completely consider the transport processes particularly at the liquid/gas interfaces, the concentration conditions would have to be measured for each nuclide at the individual points.

For the theoretical model description of nuclide transport, according to the foregoing, it is necessary to know two distribution coefficients which determine the activity flow. The one coefficient f indicates to what extents the mass flow consists of water and steam, i.e., how the mass is distributed between the liquid and gaseous phases. The second coefficient α indicates to what extents the activity is transported in the water or steam, i.e., how the activity is distributed between the two phases. f is heavily dependent on pressure, temperature and technical measures, for instance those taken to reduce the residual moisture or so-called steam content by the water separator; for this purpose the chemical or physical properties of solubility are decisive for α. The coefficients f and α can be defined in a way which is compatible with the values DF, VTF and EF previously introduced for the SGs.

In the simulation model, the secondary loop is subdivided into 13 points. To model the mass flow conditions, at least 22 mass flows are defined among the 13 points. Now a decontamination factor DF, which is a function of the mass and activity distribution factors f and α, must be assigned to each of these mass flows. For each mass flow $\dot{m}_i$ and each nuclide N_j, a decontamination factor DF_{ij} is thus to be stated which can be determined from the pair of values (f_{ij}, α_{ij}) and which represents a dimension figure for

the activity transfer between the points which mass flow $\dot{m}_i$ connects.

The functional relationship between the values f and α as well as the derived value DF is derived in the following section.

2.2.2. Transport Model for the Decontamination Factor DF as a Function of the Phase Distribution Factors f and α

The decontamination factor DF (also called decofactor for short) normally describes the ratio between the mean specific activity of a nuclide in the one (usually aqueous) phase $\bar{c}_1$ to the adjacent (usually vaporous) phase $\bar{c}_2$:

$$DF = \frac{\bar{c}_1}{\bar{c}_2} \qquad (1)$$

Accordingly, the decofactor is a measure of the transport resistance which an interface represents for a certain nuclide.

Let there be an arbitrary point i in the secondary loop. Let this point be represented by a volume with the mass content M_i and activity A_{ij} of nuclide j; at this point the following mean specific activity will thus be present:

$$\bar{c}_{ij} = \frac{A_{ij}}{M_i} \qquad (2)$$

Let the mass content M now be divided as follows:

$$M = M' + M'' \text{ with} \qquad (3)$$

$$M' = f \cdot M \text{ and } M'' = (1-f)M \qquad (3a)$$

into a liquid phase $f \cdot M$ and a gaseous phase $(1-f)M$; here the mass distribution factor f indicates the liquid portion of the total mass content.

Analogously the activity A is composed of the two activities A' in the liquid and A'' in the gaseous phases of the mass content:

$$A = A' + A''. \qquad (4)$$

Because of the difference in density between the liquid and gaseous phases, it is sound practice to relate the specific activities of the phases (stated in Ci/t). This is the same thing as the nuclide concentration in water or steam. For the sake of simplicity, we will therefore use the term "concentration" below. In addition, these concentrations will pertain to an arbitrary, but fixed nuclide.

If c' and c'' designate the specific activities or concentrations in the liquid and gaseous phases of the total mass content M, then according to (3a) and (4), the following is true of the total activity:

$$A = c' \cdot M \cdot f + c'' \cdot M \cdot (1-f) \qquad (5)$$

By analogy with (2), the following is obtained for the mean concentration $\bar{c}$ of the mass content:

$$\frac{A}{M} = \bar{c} = c' \cdot f + c'' \cdot (1-f) \tag{6}$$

If (6) is true for an arbitrary point i, then it is also true for the adjacent point j. In particular, because of the fact that M and A can always be differentiated, formula (6) can also be derived for the activity flow A and the mass flow $\dot{m}$.

From (1) and (6) it then follows for the decofactor DF that:

$$DF = \frac{c_1' f_1 + c_1''(1-f_1)}{c_2' f_2 + c_2''(1-f_2)} \tag{7}$$

The mass distribution factor f indicates how the liquid and gaseous phases are distributed over a pre-established mass (mass flow). Thus, it is not yet determined how the activity is distributed between the phases. It is a good idea for the activity distribution factor α to be defined as follows:

$$\alpha = \frac{c'}{c''} \tag{8}$$

Thus the factor α indicates the ratio of the specific activity in the liquid phase to the specific activity in the gaseous phase. α is a complicated function of, among other things, temperature, pressure, concentrations and enthalpies of all other nuclides in the water and steam phases.

With the aid of (8), (7) is transformed into:

$$DF = \frac{c_1' \left[f_1 + \frac{1}{\alpha_1}(1-f_1)\right]}{c_2' \left[f_2 + \frac{1}{\alpha_2}(1-f_2)\right]} \tag{9}$$

or into a reciprocal presentation which is constant for all values of α:

$$\frac{1}{DF} = \frac{c_2' \cdot \alpha_1 \left[\alpha_2 \cdot f_2 + 1 - f_2\right]}{c_1' \cdot \alpha_2 \left[\alpha_1 \cdot f_1 + 1 - f_1\right]} \tag{10}$$

or

$$\frac{1}{DF} = \frac{c_2' \alpha_1 \left[f_2(\alpha_2 - 1) + 1\right]}{c_1' \alpha_2 \left[f_1(\alpha_1 - 1) + 1\right]} \tag{10a}$$

-- Discussion of the Decofactor in the Form of (10) or (10a)

If for the sake of simplicity we ignore the phase distribution at one point and assume a homogeneous distribution of the activity A over the mass content M of the point, then with (2) it follows from (1) and (6) that:

$$\frac{1}{DF} = \frac{\bar{c}_2}{\bar{c}_1} = \frac{c_2' \cdot f}{\bar{c}_1} + \frac{c_2''(1-f)}{\bar{c}_1} \qquad (11)$$

(the subscripts 1 and 2 are used to distinguish point 1 and mass flow 2). The summands on the right-hand side of (11) can be interpreted as follows:

$$EF = \frac{\bar{c}_1}{c_2' f} = \text{entrainment factor} \qquad (11a)$$

and

$$VTF = \frac{\bar{c}_1}{c_2''(1-f)} = \text{vapor transfer factor} \qquad (11b)$$

from which we find for the decofactor DF the formula which is known from the literature [3] and applies to the SG

$$\frac{1}{DF} = \frac{1}{EF} + \frac{1}{VTF} \qquad (11c)$$

If, in addition, we assume that the concentration in the liquid phase of the mass flow c_2' is equal to the mean initial concentration at point $\bar{c}_1$, an assumption which is approximately valid for salts which are little-soluble in steam without re-evaporation of droplets (in the main steam) [3], [4], [5], for the entrainment factor EF the following is true:

$$EF = \frac{1}{f} \qquad (12)$$

and

$$\frac{1}{DF} = \frac{1}{VTF} + f \qquad (12a)$$

Equation (12a) can now be verified for sodium which is little soluble in steam and is transported virtually exclusively in the liquid phase, i.e., with the residual moisture of the main steam: with the literature values [6] for the VTF of sodium of $VTF > 10^6$, it follows that:

$$\frac{1}{DF} \approx f \qquad (13)$$

for which reason the element sodium, which is measurable in traces (detection limit approximately 0.1 ppb) is used to "measure" the residual moisture in the main steam [7], [8].

If we now take into account the activity distribution over the water phases by using (8), it follows directly from (11b) and (12a) that:

$$\frac{1}{DF} = \frac{1}{\alpha}(1-f) + f \qquad (14)$$

and for sufficiently small f, there follows the simple expression:

$$\frac{1}{DF} = f + \frac{1}{\alpha} \qquad (14a)$$

The simplifying assumptions which lead to (14a) are summarized below:

1. Let the activity at that point be distributed homogeneously over the mass.

2. Let the concentration in the residual moisture be equal to that at that point.

3. Let the concentration ratio α in the areas under consideration be constant and independent of the concentrations.

4. Let the residual moisture be low; $f << 1$.

These assumptions are only conditionally valid. Heavily dissociated salts in the SG and main steam with low residual moisture correspond to these assumptions.

If formula (14a) still describes the conditions in the SG correctly to some extent, then it provides completely irrelevant values for, for instance, the high-pressure turbine. However, (14a) provides false values even for gases in the SG.

If accordingly the water content of the point in question is also taken into account, then formula (10) or (10a) again results. These equations (10) and (10a) are immediately transformed into (14) on the basis of assumptions 1 and 2, i.e., with $c_1 = 1$ and $c_1' = c_2'$.

As well, on the assumption that α is dependent only on pressure, temperature, the solvent and the nuclide under consideration and that the differences between the pressure and temperature at that point and those of the mass flow leaving that point are slight, the following is approximately true:

$$\alpha_1 = \alpha_2 \qquad (15)$$

and thus:

$$\frac{1}{DF} = \frac{c_2'}{c_1'} \cdot \frac{f_2 - \frac{1}{1-\alpha}}{f_1 - \frac{1}{1-\alpha}} \qquad (16)$$

From approximation (15) it also follows from (8) that:

$$\frac{c_1'}{c_1''} = \frac{c_2'}{c_2''} \qquad (17)$$

This means, however, that:

$$\frac{c_1'}{c_2'} = \frac{c_1''}{c_2''} \qquad (17a)$$

Finally, however, from (15) there also follows the formula used in the program for the decontamination factor DF_{ij} for point i and the mass flow leaving it j:

$$DF_{ij} = \frac{\alpha \cdot f_i + 1 - f_i}{\alpha \cdot f_{ij} + 1 - f_{ij}} \qquad (18)$$

A detailed description of the value ranges of all decofactors which occur is given in Section 2.5.

2.2.3. Bibliography for 2.2

1. CRC - Handbook of Chemistry and Physics, 55th Edition, Cleveland, Ohio, 1973.

2. D'Ans, Lax, "Taschenbuch fuer Chemiker und Physiker" [Handbook for Chemists and Physicists], Vol. I: Macroscopic Physiochemical Properties, Berlin, Heidelberg, New York, 1967.

3. Hood, G. W., "Steam Side Iodine Transport Study at Point Beach Unit No. 1 of Wisconsin Electric Power Company WCAP-8215, 1973.

4. Class, G., "The Entrainment of Water by Gases or Steam", MITTEILUNGEN DER VGB, Vol. 96, pp. 179-184, 1965.

5. The Hannover Technical Supervisory Association, Response to Items CV to CIX of the Grounds for Complaint in the Administrative Court Proceedings Concerning the Grohnde NPP, KWG 806 2/1, 1978.

6. Jones, O., "Transfer of Chemicals in PWR Systems: Secondary Side; Water Chemistry of Nuclear Power Company", Wisconsin Electric Power Company WCAP-8215, 1973.

7. Doerr, A. and F. Thunkert, "Experience with the Determination of Sodium in the ppb Range", VGB-KRAFTWERKSTECHNIK, 56th year, No. 3, pp. 160-166, 1976.

8. Dibelius, G., A. Doerr and K. Ederhof et al., "Experience with the Determination of Steam Moisture in the Acceptance Tests at the Biblis NPP", KRAFTWERKSTECHNIK 9/1977.

In the following Sections 2.3 to 2.6, the input parameters of the computer program are analyzed and the corresponding values or value ranges are determined. In discussing the individual parameters, problems are discussed at many points which have either been ignored or inadequately discussed in the reactor safety debate.

2.3. Primary Loop Activities

- Introduction

The activity inventory of the fuel elements and the release rates are the initial values for emission calculations for normal operation as well as for incidents and accidents. The larger the nuclide-specific initial activity, the larger the emission ultimately will be since all barriers (fuel element cladding, cold walls, filters, etc.) will always hold up only a certain portion. The selection of the nuclides is decisive. If a nuclide is forgotten from the very start, then later in the emission calculations and in the calculation of the radiologic consequences it can no longer play a role -- an underestimate is made.

It is surprising to note that different authors or institutions consider different nuclides "important". Either no statements at all or only very general statements are made about the selection criteria which lead to the evaluation "important nuclide" or "insignificant nuclide".

In contrast, however, on the basis of the selected nuclides exact values are calculated for the emission and immission burden of nuclear plants. This applies both to work from the Reactor Safety Company (GRS) and the Power Plant Union (KWU).

A precise examination of the nuclide selection in the safety reports [7], [8] and in the German Risk Study [5] showed that even important nuclides of the essential trace elements which are indispensable for metabolism are not taken into account.

As presented in the following sections, criteria were developed for nuclide selection and an appropriate data file was compiled. Literature analyses clarified the following questions:

-- What nuclides have been detected to date in the primary loops of PWRs?

-- How high are the nuclide-specific activities? Under what conditions?

-- What values do nuclide-specific spiking factors have during load ramps or shutdown processes?

2.3.1. Summary of the Important Nuclides and Determination of Primary Loop Activities on the Basis of the Measured Data

In the operation of NPPs, experience teaches that defects occur in fuel element cladding, causing radioactive fission products (for instance, transuraniums) to get into the primary coolant. The greater the fuel element damage and the larger the transients (for instance, in shutting down the reactor or in the event of incidents or accidents), the larger is the proportion of the radioactive inventory of the fuel elements which is initially released into the primary coolant. In addition, the high neutron flux densities create activated corrosion products from the enclosures of the primary coolant. A large portion of the metallic activated corrosion products are deposited on the surfaces of the enclosures (in particular the fuel element surfaces). It is also true for the corrosion products that transients which occur initially

release a more or less large part of the deposited nuclides into the primary water.

2.3.1.1. The Selection Criterion for Important Nuclides

A NPP creates over 500 nuclides which have very different specific activities and very different half-lives. The selection criterion adopted was the absolute number N of nuclei of a radionuclide in the fuel elements:

$$N = A \cdot \tau \text{ mit } N \geq 3{,}7 \cdot 10^{16} \ *$$

where A = Activity in the fuel elements in Ci/t;

τ = Mean lifetime in sec;

$\tau = \frac{T_{1/2}}{\ln 2}$ with $T_{1/2}$ = half-life.

All nuclides for which $N \geq 3.7 \cdot 10^{16}$ are listed as important nuclides and the corresponding primary loop activities are determined. The lower boundary of $3.7 \cdot 10^{16}$ was selected such that this criterion accomplishes the following:

1). All nuclides which are defined by the KWU [7], [8] and GRS [5] as important are covered by this criterion.

2). In addition, nuclides with a short half-life (in the second or minute range) are considered if their specific activities in the fuel elements are correspondingly large so that the product is $A \cdot \tau \geq 3.7 \cdot 10^{16}$.

These short-lived nuclides must not be neglected since in most cases they are transformed in several stages into longer-lived ones and thus, in the case of emissions, into nuclides which are relevant for incident to accident induced damage.

Let us list the following as examples:

$$\text{Kr } 87 \xrightarrow[\beta^-]{76\text{min}} \text{Rb } 87 \xrightarrow[\beta^-]{4{,}7 \times 10^{10}\text{a}}$$

$$\text{Kr } 90 \xrightarrow[\beta^-]{32{,}3\text{sec}} \text{Rb } 90 \xrightarrow[\beta^-]{2{,}6\text{min}} \text{Sr } 90 \xrightarrow[\beta^-]{28{,}5\text{a}}$$

$$\text{Kr } 93 \xrightarrow[\beta^-]{1{,}29\text{sec}} \text{Rb } 93 \xrightarrow[\beta^-]{5{,}8\text{sec}} \text{Sr } 93 \xrightarrow[\beta^-]{7{,}45\text{min}} \text{Y } 93 \xrightarrow[\beta^-]{10{,}1\text{h}}$$

$$\text{Zr } 93 \xrightarrow[\beta^-]{1{,}5 \times 10^{6}\text{a}}$$

*The factor 3.7 results from the definition of the unit Curie $3.7 \cdot 10^{10}$ sec^{-1}.

$$\text{Nb } 101 \xrightarrow[\beta^-]{7{,}15\text{min}} \text{Mo } 101 \xrightarrow[\beta^-]{14{,}6\text{min}} \text{Tc } 101 \xrightarrow[\beta^-]{14\text{min}}$$

$$\text{Xe } 137 \xrightarrow[\beta^-]{3{,}83\text{min}} \text{Cs } 137 \xrightarrow[\beta^-]{30{,}1\text{a}}$$

$$\text{Xe } 140 \xrightarrow[\beta^-]{13{,}5\text{sec}} \text{Cs } 140 \xrightarrow[\beta^-]{64\text{sec}} \text{Ba } 140 \xrightarrow[\beta^-]{12{,}79\text{d}} \text{La} \xrightarrow[\beta^-]{40{,}2\text{h}}$$

$$\text{Ce } 149 \xrightarrow[\beta^-]{5\text{s}} \text{Pr } 149 \xrightarrow[\beta^-]{28\text{s}} \text{Nd } 149 \xrightarrow[\beta^-]{1{,}73\text{h}} \text{Pm } 149 \xrightarrow[\beta^-]{53\text{h}} .$$

Just these few examples show that ignoring the short-lived nuclides necessarily leads to an underestimation of the consequences of emissions.

3). All nuclides with a low specific activity are considered if their half-lives are so long that the product is $A \cdot \tau \geq 3.7 \cdot 10^{16}$. This ensures that, for instance, actinides with, in some cases, half-lives of several thousand years are taken into account. Even with low specific activity, these nuclides are of particular importance in the area of long-term consequences because of their half-lives, which last many generations in the case of emissions.

4). In addition, the essential (E) or potentially essential (PE) trace elements [1] which are of particular importance for the isotopes of warm-blooded animals are considered by the criterion $A \cdot \tau \geq 3.7 \cdot 10^{16}$.

An additional evaluation [of the damage] caused, for instance, by ingestion toxicity or inhalation toxicity was omitted since the data base on which these statements are founded is the subject of intense scientific dispute (compare [15]) and also because no inhalation or ingestion factors are described in the literature for some of these nuclides.

2.3.1.2. Explanation of the Calculation of Primary Loop Activities and Spiking Factors on the Basis of Measured Data

Table 2.3.1.2.1 summarizes the corresponding data. The calculation of the individual data is described in the sequence of the individual columns.

Column 1: Designation of the nuclide

Column 2: A for activation product
S for fission product

Column 3: Half-lives in [9]: s = seconds
m = minutes
h = hours
d = days
a = years

Column 4: The inventory of radioactive substances in the reactor core was determined with the ORIGEN computer code [10]* and was stated at the end of the third cycle for the KWU Biblis B reactor. Since Biblis A, Phillipsburg II, Unterweser, the Sued NPP, Grohnde, Neckarwestheim II and Grafenrheinfeld are almost identical [to this reactor] in their thermal power, it can be assumed that the core inventories of these plants are approximately the same.

Column 5: Measured or calculated primary loop activities. Curies per ton of primary coolant are indicated. If the same isotope occurs both as a fission product = S and an activation product = A (compare column 2), then the primary loop activities were consolidated.

Column 6: Classification of the primary loop (PL) activity:

m = measured
i = isotope
s = similar
v = volatile
sst = solid state
a = actinide.

The classifications are explained below:

Detailed definitions: m = measured; these measured values were taken from the literature, in most cases from German PWRs. If several [different] measured values were found, then to stay on the conservative side the highest value was selected. In this process it was ensured that these were not values which were created by spikes. A comparison with the design values of the KWU (compare column 10) shows good agreement.

i, s, v, sst and a mean that these values were calculated on the basis of nuclides which were as similar as possible and for which measured values are known. Since measured primary concentrations are taken as the point of departure, in principle the mass, clean-up and mean water exchange of the primary coolant as well as leakage into the sump are taken into account. The calculation was performed by means of the equation:

$$C_{iPK} = \frac{C_{jPK}}{C_{jKI}} \cdot C_{iKI}$$

*The computations with the above-indicated program were carried out by Gerald Kirchner, Certified Physicist of the University of Bremen.

C_{iPK} = Required primary loop concentration of the i-th nuclide.

C_{iKI} = Core inventory of the i-th nuclide calculated with the ORIGEN program, in Ci/t of heavy metal;*

C_{jPK} = Measured primary loop concentration of the j-th nuclide in Ci/t of coolant which is as similar as possible to the i-th nuclide in terms of physiochemical behavior;

C_{jKI} = Core inventory of the j-th nuclide calculated with the ORIGEN program, in Ci/t of heavy metal;

s = Similar; elements i and j are nuclides of the same main group or the same adjacent group in the periodic table.

v = Volatile; elements i and j are nuclides from the group "solids which are volatile under 1200°" (according to [11]);

sst = Solid state; elements i and j are nuclides from the group "other solids" (according to [11]);

a = Actinide; elements i and j are nuclides from the group "actinides" (according to [11]).

Column 7: Essential (E) and potentially essential trace elements (EP) [sic] according to K. Betke and F. Bidlingmaier [1].

Column 8: Statement of the upper spiking factor SF (compare Figure 2.3.1.2) as well as of a mean spiking factor SF_m. The mean spiking factor SF_m is average from the values of several publications.

Column 9: Total spike time T_{tot}, as well as the mean spiking time T_m.

Using the example of the work of W. Chubb [29] it is explained how the values in column 8 (spiking factors) and column 9 (spiking times) were obtained [29] (Figure 2.3.1.2).

The work of W. Chubb [29] was used since it is one of the few of the more than 20 works studied which gives data which provide a detailed picture of the variation of spike activity because of its relatively extensive activity measurements. In most of the other works, the differentiated structure of the spiking variation is not recognizable. Then the indicated spiking factor is equal to the maximum spiking factor which is determined from the maximum of the measured spike activity and not from the maximum of the averaged spike activity.

The so-called spiking concentration is the maximum specific activity A_s in the primary loop caused by a load change or a transient of the reactor. (Depending on the degree of detail of the data, A_s is designated as the maximum averaged specific activity, i.e., the "upper spiking activity").

The ratio of the maximum specific activities A_s to the specific activity A_o prior to the peak yields the spiking factor SF:

$$SF = \frac{A_s}{A_o}$$

i.e., two characteristic time indications are obtained for the spike (compare Figure 2.3.1.2):

-- 1. The total spike time T_{tot} with $T_{tot} = T_3 - T_o$. T_{tot} is the time from the beginning of the spike-induced activity peak T_o to its attenuation back to the initial activity A_o at time T_3.

-- 2. The mean spike time T_m with $T_m = T_2 - T_1$. T_m is the spike time which would result if the spiking activity A_S were constant during time $T_2 - T_1$. Thus T_m is determined in such a way that:

$$\int_{T_0}^{T_3} [A(t) - A_o]\, dt = T_m\, (A_S - A_o)$$

A(t) = Time-dependent activity variation.

Column 10: Design activities of the KWU in the reactor cooling system (primary loop activities) in Curies per ton of coolant.

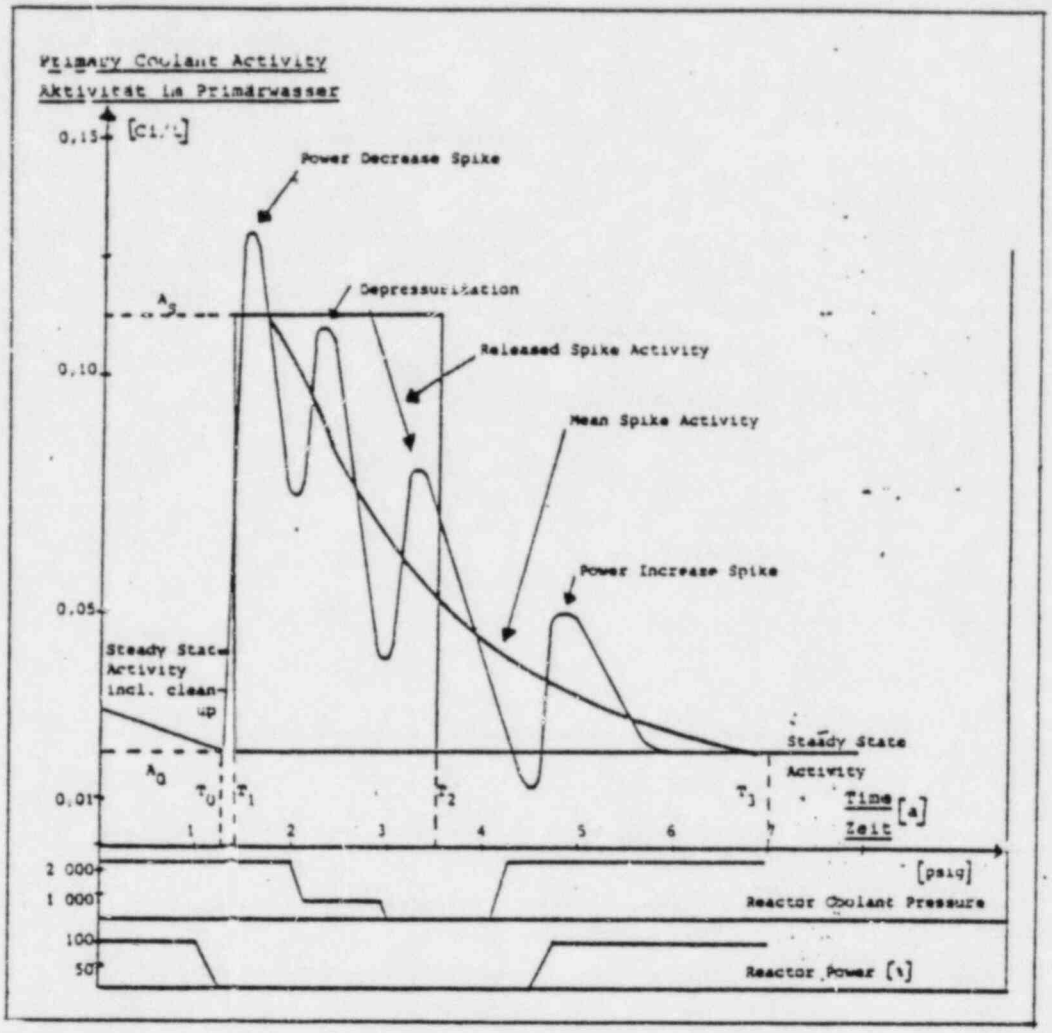


Figure 2.3.1.2. Iodine Spiking Behavior in a PWR According to W. Chubb [29].

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
H 3	S	12.35 a	3.56 (-2)						5.0 (-1)
H 3	A	12.35 a	5.42 (+2)						
Be 10	A	1.6 (+6)a	2.01 (-7)	1.0 (-8)		EP			
C 14	A	5,736 a	2.94 (-4)	8.8 (-12)	s				
Na 24	A	15.03 h	4.18	xxx					
Al 28	A	224.6 m	8.43 (+2)	xxx	s	EP			
Si 31	A	2.62 h	1.34 (+2)	4.0 (-6)	s	E			
P 32	A	14.3 d	2.08 (+2)	1.1 (-3)	s				
P 33	A	25.3 d	5.19 (+2)	2.9 (-3)	s				
S 35	A	87.5 d	3.84	1.9 (-7)	s				
Cl 36	A	300,000 a	6.68 (-7)	1.7 (-14)	s				
Ca 45	A	163 d	1.61 (-1)	4.8 (-9)	s				
Sc 46	A	84 d	2.14 (+1)	6.4 (-7)	s	EP			
Sc 47	A	3.42 d	3.58 (+1)	1.0 (-6)	s	EP			
Sc 48	A	43.67 h	3.24	xxx		EP			
V 52	A	3.75 m	2.14 (+3)	xxx		E			
Cr 51	A	27.7 d	1.64 (+5)	8 (-3)	m	E	10 (7.5)	72 h (48 h)	2.0 (-3)
Cr 55	A	3.6 m	2.29 (+3)	xxx		E			
Mn 54	A	312 d	5.13 (+3)	1 (-3)	m	E	37.5 (22)	72 h (48 h)	2.0 (-4)
Mn 56	A	2.58 h	3.18 (+5)	6 (-4)	m	E	100		
Fe 55	A	2.7 a	3.92 (+4)	8.7 (-3)	i	E			
Fe 59	A	44.6 d	3.06 (+3)	7 (-4)	m	E	20		2.0 (-4)
Co 58	A	70.78 d	1.86 (+4)	4.4 (-3)	m	E	10000 (2200)	400 h (100 h)	2.0 (-3)
Co 60m	A	10.5 m	3.20 (+4)	6.2 (-3)	i	E			
Co 60	A	5.272 a	1.46 (+4)	2 (-3)	m	E	500 (94)	400 h (100 h)	6.0 (-4)
Co 61	A	1.6 h	3.98 (+2)	7.7 (-5)	i	E			
Ni 59	A	75,000 a	1.10 (+1)	2.3 (-6)	s	E			

5 (-4) = 5×10^{-4} xxx Nuclides which lie just below the criterion limit of $N \geq 3.7 \cdot 10^{15}$.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Ni 63	A	100 a	1.69 (+3)	3.5 (-4)	s	E			
Ni 65	A	2.52 h	3.16 (+3)	6.6 (-4)	s	E			
Cu 64	A	12.7 h	4.61 (+2)	1.4 (-3)	sst	E			
Zn 65	A	244 d	2.56 (-1)	6.8 (-5)	m	E			
Zn 72	S	46.5 h	2.20 (+1)	5.8 (-3)	i	E			
Ga 72	S	14.1 h	2.23 (+1)	6.7 (-7)	sst	EP			
Ga 73	S	4.8 h	5.45 (+1)	1.6 (-6)	sst	EP			
Ge 75	S	83 m	3.59 (+2)	1.1 (-5)	sst	EP			
Ge 77	S	11.3 h	8.53 (+2)	2.6 (-5)	sst	EP			
Ge 77m	S	54 s	1.70 (+3)	xxx		EP			
Ge 78	S	88 m	5.60 (+3)	1.7 (-4)	sst	EP			
Ge 79	S	42 s	1.14 (+4)	xxx		EP			
Ge 80	S	24.5 s	2.37 (+4)	xxx		EP			
Ge 81	S	10.1 s	2.54 (+4)	xxx		EP			
Ge 82	S	4.6 s	1.98 (+4)	xxx		EP			
As 76	S	26.4 h	2.62 (+1)	1.4 (-4)	s				
As 77	S	38.8 h	2.21 (+3)	1.2 (-2)	s				
As 78	S	1.5 h	5.84 (+3)	3.2 (-2)	s				
As 79	S	8.2 m	1.40 (+4)	7.7 (-2)	s				
As 80	S	15.2 s	3.33 (+4)	xxx					
As 81	S	34 s	5.03 (+4)	2.8 (-1)	s				
As 83	S	13.3 s	6.37 (+4)	3.5 (-1)	s				
As 84	S	5.3 s	4.87 (+4)	xxx					
Se 79	S	65 000 a	3.90 (-1)	1.9 (-8)	c	E			
Se 79m	S	3.9 m	1.40 (+4)	6.9 (-4)	v	E			

5 (-4) = 5×10^{-4} xxx Nuclides which lie just below the criterion limit of $N \geq 3.7 \cdot 10^{16}$.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Se 81	S	18 m	5.53 (+4)	2.7 (-3)	v	E			
Se 81m	S	57.3 m	1.67 (+3)	8.2 (-5)	v	E			
Se 83	S	22.4 m	4.48 (+4)	2.2 (-3)	v	E			
Se 83m	S	69 s	6.44 (+4)	3.2 (-3)	v	E			
Se 84	S	3.1 m	1.78 (+5)	8.7 (-3)	v	E			
Se 85	S	33 s	1.08 (+5)	5.3 (-3)	v	E			
Se 86	S	16.1 s	1.92 (+5)	9.4 (-3)	v	E			
Se 87	S	5.6 s	1.70 (+5)	8.3 (-3)	v	E			
Se 88	S	1.5 s	5.43 (+4)	xxx		E			
Br 82	S	35.34 h	4.21 (+3)	8.8 (-4)	s	EP			
Br 82m	S	6.1 m	3.35 (+3)	7.0 (-4)	s	EP			
Br 84±	S	31.8 m	1.87 (+5)	3.9 (-2)	s	EP			
Br 84m	S	6.0 m	9.46 (+3)	2.0 (-3)	s	EP			
Br 85	S	2.87 m	2.41 (+5)	5.1 (-2)	s	EP			
Br 86	S	54 s	1.63 (+5)	3.4 (-2)	s	EP			
Br 86m	S	4.5 s	1.64 (+5)	3.4 (-2)	s	EP			
Br 87	S	55.7 s	3.83 (+5)	8.0 (-2)	s	EP			
Br 88	S	16.2 s	3.94 (+5)	8.3 (-2)	s	EP			
Br 89	S	4.5 s	2.61 (+5)	5.21 (-2)	s	EP			
Br 90	S	1.63 s	1.63 (+5)	xxx		EP			
Kr 81	S	210 000 a	5.63 (-7)	1.8 (-13)	i				
Kr 83m	S	1.83 h	1.13 (+5)	3.8 (-2)	i				
Kr 85	S	10.76 a	9.98 (+3)	3.4 (-3)	i				4.9 (-1)
Kr 85m	S	4.48 h	2.45 (+5)	2.5 (-1)	m		30		1.5 (0)
Kr 87	S	76.3 m	4.44 (+5)	1.5 (-1)	m				1.7 (0)

5 (-4) = 5×10^{-4} xxx Nuclides which lie just below the criterion limit of $N \geq 3.7 \cdot 10^{16}$.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Kr 88	S	2.8 h	6.34 (+5)	2.9 (-1)	m				3.4 (0)
Kr 89	S	3.18 m	7.71 (+5)	2.6 (-1)	i				
Kr 90	S	32.3 s	7.56 (+5)	2.6 (-1)	i				
Kr 91	S	8.6 s	5.44 (+5)	1.8 (-1)	i				
Kr 92	S	1.84 s	2.59 (+5)	xxx					
Kr 93	S	1.29 s	9.15 (+4)	xxx					
Rb 86	S	18.7 d	2.35 (+3)	1.2 (-4)	t, s	EP			
Rb 87	S	44.7 (+9)a	2.21 (-5)	1.10 (-13)	t, s	EP			
Rb 88	S	17.8 m	6.47 (+5)	3.1 (-2)*	t, s				2.4 (0)
Rb 89	S	15.2 m	8.37 (+5)	4.1 (-2)	t, s				
Rb 90	S	2.6 m	8.11 (+5)	4.0 (-2)	t, s				
Rb 90m	S	4.3 m	2.05 (+5)	1.0 (-2)	t, s				
Rb 91	S	58 s	1.02 (+6)	5.0 (-2)	t, s				
Rb 92	S	4.5 s	8.75 (+5)	4.3 (-2)	t, s				
Rb 93	S	5.8 s	6.44 (+5)	3.2 (-2)	t, s				
Rb 94	S	2.69 s	3.20 (+5)	1.6 (-2)	t, s				
Sr 87m	S	2.81 h	9.48	xxx					
Sr 89	A	50.5 d	5.17 (+1)						
Sr 89	S	50.5 d	8.81 (+5)	2.6 (-2)	sst				
Sr 90	A	28.5 a	1.03 (-3)	3.11 (-11)	sst				9.6 (-5)
Sr 90	S	28.5 a	7.90 (+4)	2.4 (-3)	sst				9.6 (-5)
Sr 91	A	9.5 h	9.98	xxx					
Sr 91	S	9.5 h	1.10 (+6)	3.3 (-2)	sst				
Sr 92	S	2.71 h	1.20 (+6)	3.6 (-2)	sst				
Sr 93	S	7.45 m	1.38 (+6)	4.1 (-2)	sst				

*Plus 2.3 Ci/t (decay product of Kr 85).

5 (-4) = 5×10^{-4} xxx Nuclides which lie just below the criterion limit of $N \geq 3.7 \cdot 10^{16}$.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Sr 94	S	74 s	1.36 (+6)	4.0 (-2)	sst				
Sr 95	S	24.4 s	1.19 (+6)	3.6 (-2)	sst				
Sr 96	S	1 s	7.92 (+5)	2.4 (-2)	sst				
Sr 97	S	0.2 s	3.88 (+5)	xxx					
Sr 98	S	0.6 s	1.45 (+5)	xxx					
Y 89m	S	16 s	7.16 (-1)	xxx					
Y 90	S	64.1 h	8.30 (+4)	2.5 (-3)	sst				
Y 90m	S	3.19 h	2.34 (+1)	xxx					
Y 91	A	58.5 d	1.37 (+2)						
Y 91	S	58.5 d	1.15 (+6)	3.5 (-2)	sst				
Y 91m	S	49.7 m	6.41 (+5)	1.9 (-2)	sst				
Y91m	A	49.7 m	5.89						
Y 92	S	3.54 h	1.21 (+6)	3.6 (-2)	sst				
Y 93	S	10.1 h	1.42 (+6)	4.3 (-2)	sst				
Y 94	S	19 m	1.52 (+6)	4.6 (-2)	sst				
Y 95	S	10.3 m	1.60 (+6)	4.8 (-2)	sst				
Y 96	S	9.3 s	1.59 (+6)	4.5 (-2)	sst				
Y 97	S	1.1 s	1.26 (+6)	3.8 (-2)	sst				
Y 98	S	1 s	8.55 (+5)	2.6 (-2)	sst				
Zr 93	A	1.5 (+6)a	1.19 (=1)			EP			
Zr 93	S	1.5 (+6)a	1.87	2.3 (-11)	i	EP			
Zr 95	A	64 d	4.65 (+4)			EP			1 (-4)
Zr 95	S	64 d	1.66 (+6)	3.0 (-5)	m	EP			1 (-4)
Zr 97	S	16.8 h	y.70 (+6)	2.2 (-5)	m	EP			
Zr 98	S	30.7 s	1.67 (+6)	5.3 (-5)	i	EP			

5 (-4) = 5×10^{-4} xxx Nuclides which lie just below the criterion limit of $N \geq 3.7 \cdot 10^{16}$.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classifi-, cation	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Zr 99	S	2.35 s	1.66 (+6)	5.3 (-5)	i	EP			
Zr 100	S	7.1 s	1.49 (+6)	4.7 (-5)	i	EP			
Zr 101	S	2 s	8.84 (+5)	2.8 (-5)	i	EP			
Zr 102	S	2.9 s	4.66 (+5)	1.5 (-5)	i	EP			
Nb 93m	A	13.6 a	9.20 (-3)			EP			
Nb 93m	S	1.36 a	1.32 (-1)	4.9 (-9)	sst	EP			
Nb 94	A	20 000 a	2.11 (-3)	6.3 (-11)	sst	EP			
Nb 94	S	20 000 a	1.75 (-4)			EP			
Nb 95	A	35.15 d	4.77 (+4)			EP			
Nb 95	S	35.15 d	1.65 (+6)	5.0 (-2)	sst	EP			
Nb 95m	S	86.6 h	1.29 (+4)	3.6 (-4)	sst	EP			
Nb 96	A	23.4 h	3.80 (+1)			EP			
Nb 96	S	23.4 h	2.06 (+3)	6.2 (-5)	sst	EP			
Nb 97	S	74 m	1.72 (+6)	5.2 (-2)	sst	EP			
Nb 97m	S	53 s	1.61 (+6)	4.8 (-2)	sst	EP			
Nb 98	S	51 m	1.71 (+6)	5.1 (-2)	sst	EP			
Nb 101	S	7.1 s	1.57 (+6)	4.7 (-2)	sst	EP			
Nb 102	S	4.3 s	1.29 (+6)	3.9 (-2)	sst	EP			
Mo 93	A	3 500 s	2.79 (-2)	1.4 (-9)	s	E			
Mo 99	A	66 h	2.79 (+3)			E			
Mo 99	S	66 h	9.84 (+6)	9.0 (-2)	s	E			
Mo 101	S	14.6 m	1.70 (+6)	7.9 (-2)	s	E			
Mo 102	S	11.5 m	1.62 (+6)	8.1 (-2)	s	E			
Mo 103	S	62 s	1.66 (+6)	8.1 (-2)	s	E			
Mo 104	S	1.1 m	1.25 (+6)	6.1 (-2)	s	E			

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Mo 105	S	42 s	8.86 (+5)	4.3 (-2)	s	E			
Mo 106	S	10 s	3.90 (+5)	1.9 (-2)	s	E			
Tc 99	S	210 000a	1.31 (+1)	3.9 (-7)	s				
Tc 99m	A	6 h	2.43 (+3)						
Tc 99m	S	6 h	1.62 (+6)	4.9 (-2)	s				
Tc 99	A	210 000a	1.66 (-2)	4.8 (-10)	s				
Tc 100	S	15.8 s	6.47 (+5)	1.9 (-2)	s				
Tc 101	S	14 m	1.70 (+6)	5.0 (-2)	s				
Tc 102	S	5.3 s	1.63 (+6)	4.9 (-2)	s				
Tc 103	S	50 s	1.69 (+6)	5.1 (-2)	s				
Tc 104	S	18 m	1.36 (+6)	4.1 (-2)	s				
Tc 105	S	7.6 m	1.15 (+6)	3.5 (-2)	s				
Tc 106	S	36 s	7.19 (+5)	2.2 (-2)	s				
Tc 107	S	21 s	3.09 (+5)	9.3 (-3)	s				
Tc 108	S	5 s	2.06 (+5)	6.2 (-3)	s				
Ru 103	S	39.35 d	1.70 (+6)	3.6 (-2)	sst	EP			
Ru 105	S	4.44 h	1.17 (+6)	2.6 (-2)	sst	EP			
Ru 106	S	368 d	5.37 (+5)	1.1 (-2)	sst	EP			
Ru 107	S	3.8 m	6.27 (+5)	1.3 (-2)	sst	EP			
Ru 108	S	4.5 m	4.17 (+5)	8.8 (-3)	sst	EP			
Ru 109	S	34.5 s	2.57 (+5)	5.4 (-3)	sst	EP			
Ru 110	S	15 s	1.00 (+5)	2.1 (-2)	sst	EP			
Rh 103m	S	56.1 m	1.53 (+6)	3.2 (-2)	sst	EP			
Rh 104	S	42 s	1.17 (+6)	2.5 (-2)	sst	EP			
Rh 105	S	35.5 h	1.05 (+6)	2.2 (-2)	sst	EP			

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Rh 105m	S	45 s	3.28 (+5)	6.9 (-3)	sst	EP			
Rh 106	S	30 s	6.23 (+5)	1.3 (-2)	sst	EP			
Rh 106m	S	2.2 h	3.56 (+4)	7.5 (-4)	sst	EP			
Rh 106	S	22 m	6.32 (+5)	1.3 (-2)	sst	EP			
Rh 109	S	80 s	2.67 (+5)	5.6 (-3)	sst	EP			
Rh 109m	S	50 s	1.33 (+5)	2.8 (-3)	sst	EP			
Rh 110	S	27.7 s	1.09 (+5)	2.3 (-3)	sst	EP			
Rh 111	S	62.7 s	5.22 (+4)	1.1 (-3)	sst	EP			
Pd 107	S	6.5 (+6)a	1.11 (-1)	2.3 (-9)	sst	EP			
Pd 109	S	13.46 h	3.39 (+5)	7.1 (-3)	sst	EP			
Pd 109m	S	4.69 m	1.35 (+5)	2.8 (-3)	sst	EP			
Pd 111	S	22 m	5.34 (+4)	1.1 (-3)	sst	EP			
Pd 111m	S	5.5 h	9.20 (+2)	1.9 (-5)	sst	EP			
Pd 112	S	20.1 h	2.17 (+4)	4.6 (-4)	sst	EP			
Pd 113	S	1.6 m	1.62 (+4)	3.4 (-4)	sst	EP			
Pd 114	S	2.4 m	1.00 (+4)	2.1 (-4)	sst	EP			
Ag 108	S	2.41 m	1.11 (-2)	3.3 (-10)	sst	EP			
Ag 109m	S	39.6 s	3.39 (+5)	1.0 (-2)	sst	EP			
Ag 110	S	24.6 s	1.91 (+5)	5.7 (-3)	sst	EP			
Ag 110m	S	250.4 d	6.27 (+3)	1.9 (-4)	sst	EP			
Ag 111	S	7.5 d	5.40 (+4)	1.6 (-3)	sst	EP			
Ag 111m	S	1.2 m	5.32 (+4)	1.6 (-3)	sst	EP			
Ag 112	S	3.12 h	2.18 (+4)	6.5 (-4)	sst	EP			
Ag 113	S	5.37 h	1.47 (+4)	4.4 (-4)	sst	EP			
Ag 115	S	2 m	6.00 (+3)	1.8 (-4)	sst	EP			

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Cd 113m	S	14.6 a	2.31 (+1)	8.9 (-7)[?]	sst	EP			
Cd 115	S	53.38 h	8.05 (+3)	3.1 (-4)	sst	EP			
Cd 115m	S	44.8 d	8.22 (+2)	3.1 (-5)	sst	EP			
Cd 117	S	2.42 h	5.10 (+3)	1.9 (-4)	sst	EP			
Cd 117m	S	3.31 h	2.83 (+3)	1.1 (-4)	sst	EP			
Cd 118	S	50.3 m	7.97 (+3)	3.0 (-4)	sst	EP			
In 114m	S	49.5 d	1.85	5.5 (-8)	sst	EP			
In 115	S	600(+12)a	7.00 (-12)	2.1 (-19)	sst	EP			
In 115m	S	4.5 h	8.05 (+3)	2.4 (-4)	sst	EP			
In 116	S	54 m	1.78 (+3)	5.3 (-5)	sst	EP			
In 117	S	38 m	4.76 (+3)	1.4 (-4)	sst	EP			
In 117m	S	1.95 h	5.99 (+3)	1.8 (-4)	sst	EP			
In 119m	S	18 m	6.05 (+3)	1.8 (-4)	sst	EP			
Sn 117m	A	14 d	1.32 (+4)	4.0 (-4)	sst	E			
Sn 117m	S	14 d	5.11 (+3)			E			
Sn 119m	A	245 d	1.11 (+4)	3.3 (-4)	sst	E			
Sn 119m	S	245 d	1.19 (+3)			E			
Sn 121	A	27 h	4.68 (+3)			E			
Sn 121	S	27 h	8.52 (+3)	2.6 (-4)	sst	E			
Sn 121m	A	50 a	6.35 (-1)	1.9 (-8)	sst	E			
Sn 121m	S	50 a	1.72 (-1)			E			
Sn 123	A	129.2 d	3.18 (+2)			E			
Sn 123	S	129.2 d	2.20 (+3)	6.6 (-5)	sst	E			
Sn 123m	S	40.1 m	7.92 (+3)	7.4 (-4)	sst	E			
Sn 125	A	9.64 d	9.74 (+1)	2.9 (-6)	sst	E			

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Sn 125	S	9.64 d	1.14 (+4)			E			
Sn 125m	A	9.5 m	3.15 (+3)			E			
Sn 125m	S	9.5 m	1.58 (+4)	4.7 (-4)	sst	E			
Sn 127	S	2.1 h	6.11 (+4)	1.8 (-3)	sst	E			
Sn 127m	S	4.4 m	3.54 (+4)	1.1 (-3)	sst	E			
Sn 128	S	59 m	1.61 (+5)	4.8 (-3)	sst	E			
Sn 130	S	3.7 m	3.42 (+5)	1.0 (-2)	sst	E			
Sn 131	S	59 s	2.78 (+5)	8.3 (-3)	sst	E			
Sn 132	S	40 s	1.46 (+5)	4.4 (-3)	sst	E			
Sb 122	S	2.7 d	1.45 (+3)	8.0 (-3)	m	EP	40 (22)	72 h (48 h)	
Sb 124	A	60.3 d	1.0 (+1)			EP	1000 (137)	72 h (48 h)	
Sb 124	S	60.3 d	1.06 (+3)	2.2 (-2)	m	EP			
Sb 125	A	2.77 a	1.30 (+3)			EP			
Sb 125	S	2.77 a	9.66 (+3)	5.3 (-2)	i	EP			
Sb 126	A	12.4 d	7.6	4.2 (-5)	i	EP			
Sb 126	S	12.4 d	7.38 (+2)	4.1 (-3)	i	EP			
Sb 127	S	3.85 d	1.05 (+5)	5.8 (-1)	i	EP			
Sb 129	S	4.32 h	3.45 (+5)	1.9	i	EP			
Sb 131	S	23 m	8.16 (+5)	4.5	i	EP			
Sb 133	S	2.3 m	4.99 (+5)	2.8	i	EP			
Te 123	S	12.4(+12)a	8.081(-13)	4.0 (-20)	v	EP			
Te 123m	S	119.7 d	1.33 (+1)	6.5 (-7)	v	EP			
Te 125m	A	58 d	4.63 (+2)	2.3 (-5)	v	EP			
Te 125m	S	58 d	2.00 (+3)	9.8 (-5)	v	EP			
Te 127	S	9.36 h	1.04 (+5)	5.1 (-3)	v	EP			

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Te 127m	S	109 d	1.33 (+4)	6.5 (-4)	v	EP			
Te 129	S	69.6 m	3.41 (+5)	1.7 (-2)	v	EP			
Te 129m	S	33.6 d	5.20 (+4)	2.5 (-3)	v	EP			
Te 131	S	25 m	8.92 (+5)	4.4 (-2)	v	EP			
Te 131m	S	30 h	1.59 (+5)	7.8 (-3)	v	EP			
Te 132	S	78 h	1.45 (+6)	7.1 (-2)	v	EP			
Te 133	S	12.5 m	1.13 (+6)	5.5 (-2)	v	EP			
Te 113m	S	55.4 m	7.37 (+5)	3.6 (-2)	v	EP			
Te 134	S	41.8 m	1.48 (+6)	7.3 (-2)	v	EP			
Te 135	S	18 s	7.46 (+5)	3.7 (-2)	v	EP			
Te 137	S	3.5 s	3.94 (+5)	1.9 (-2)	v	EP			
J 128	S	25 m	1.27 (+4)	2.6 (-3)	i	E			
J 129	S	15.7(+6)a	3.23 (-2)	6.7 (-9)	i	E			3.2 (-8)
J 130	S	12.36 h	2.54 (+4)	5.2 (-3)	i	E			
J 130m	S	9 m	1.85 (+4)	3.8 (-3)	i	E			
J 131	S	8.04 d	1.02 (+6)	2.7 (-1)	m	E	100 (52)	50 h (20 h)	9.2 (-1)
J 132	S	2.38 h	1.48 (+6)	3.0 (-1)	i	E			3.2
J 133	S	20.8 h	2.04 (+6)	3.0 (-1)	m	E	60 (18)	7d (4d)	4.8
J 134	S	52 m	2.18 (+6)	4.5 (-1)	i	E			3.9
J 134m	S	3.5 m	2.48 (+5)	5.1 (-2)	i	E			
J 135	S	6.59 h	1.89 (+6)	3.9 (-1)	i	E			4.8
J 136	S	83 s	8.23 (+5)	1.7 (-1)	i	E			
J 136m	S	46 s	5.14 (+5)	1.1 (-1)	i	E			
J 137	S	24.2 s	7.99 (+5)	1.6 (-1)	i	E			
J 138	S	6.2 s	3.82 (+5)	8.1 (-2)	i	E			

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Xe 129m	S	8.89 d	7.61	1.2 (-4)	i				
Xe 131m	S	12 d	1.03 (+4)	1.7 (-1)	i				
Xe 133	S	5.29 d	2.05 (+6)	6.0 (+1)	m		500 (160)	130 h (70 h)	20
Xe 133m	S	2.2 d	6.60 (+4)	1.1	i				4.9 (-1)
Xe 135	S	9.17 h	5.03 (+5)	2.0	m		10		8.2
Xe 135m	S	15.3 m	4.31 (+5)	7.2	i				5.1 (-1)
Xe 137	S	3.83 m	1.79 (+6)	2.9 (+1)	i				
Xe 138	S	14.1 m	1.61 (+6)	2.7 (+1)	i				1.4
Xe 139	S	39.7 s	121 (+6)	2.9 (+1)	i				
Xe 140	S	13.5 s	7.81 (+5)	1.3 (+1)	i				
Cs 134	S	2.06 a	2.52 (+5)	5.5 (-3)	m	EP	275		3.1 (-3)
Cs 134m	S	2.9 h	6.84 (+4)	3.4 (-3)	i	EP			
Cs 135	S	2 (+6) a	4.59 (-1)	2.2 (-8)	i	EP			
Cs 135m	S	53 m	7.42 (+2)	3.6 (-5)	i	EP			
Cs 136	S	13 d	3.04 (+4)	1.5 (-3)	i	EP			
Cs 137	S	30.1 a	1.12 (+5)	5.5 (-3)	m	EP	500 (390)		9.4 (-3)
Cs 138	S	32.2 m	1.71 (+6)	8.4 (-2)*	i	EP			1.3
Cs 138m	S	2.9 m	1.00 (+5)	4.9 (-3)	i	EP			
Cs 139	S	9.3 m	1.70 (+6)	8.3 (-2)	i	EP			
Cs 140	S	64 s	1.54 (+6)	7.5 (-2)	i	EP			
Cs 141	S	24.7 s	1.11 (+6)	5.4 (-2)	i	EP			
Cs 142	S	1.68 s	6.34 (+5)	3.1 (-2)	i	EP			
Ba 135m	S	28.7 h	2.19 (+2)	6.6 (=6)	sst	EP			
Ba 137m	S	2.55 m	1.06 (+5)	3.2 (-3)	sst	EP			
Ba 139	S	82.7 m	1.78 (+6)	5.3 (-2)	sst	EP			

*Plus 1.3 Ci/t (decay product of Xe 138).

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Ba 140	S	12 79 d	1.75 (+6)	5.3 (-2)	sst	EP			
Ba 141	S	18.3 m	1.64 (+6)	4.9 (-2)	sst	EP			
Ba 142	S	10.7 m	1.54 (+6)	4.6 (-2)	sst	EP			
Ba 143	S	20 s	1.33 (+6)	4.0 (-2)	sst	EP			
Ba 144	S	11.9 s	9.64 (+5)	2.9 (-2)	sst	EP			
Ba 145	S	5.6 s	4.95 (+5)	1.5 (-2)	sst	EP			
La 138	S	130 (+9)a	1.22 (-10)	3.7 (-18)	sst	EP			
La 140	S	40.2 h	1.82 (+6)	5.5 (-2)	sst	EP			
La 141	S	3.93 h	1.65 (+6)	5.0 (-2)	sst	EP			
La 142	S	92.5 m	1.58 (+6)	4.7 (-2)	sst	EP			
La 143	S	14.3 m	1.50 (+6)	4.5 (-2)	sst	EP			
La 145	S	29 s	9.38 (+5)	2.8 (-2)	sst	EP			
La 146	S	8.3 s	5.98 (+5)	1.8 (-2)	sst	EP			
La 147	S	1.6 s	2.88 (+5)			EP			
Ce 141	S	32 41 d	1	3.9 (-8)	sst	EP			
Ce 143	S	33 h	1.51 (+6)	4.5 (-2)	sst	EP			
Ce 144	S	284.8 d	120 (+6)	3.6 (-2)	sst	EP			
Ce 145	S	3 m	1.02 (+6)	3.1 (-2)	sst	EP			
Ce 146	S	13.9 m	8.14 (+5)	2.4 (-2)	sst	EP			
Ce 147	S	57 s	6.28 (+5)	1.9 (-2)	sst	EP			
Ce 149	S	5 s	2.45 (+5)	7.4 (-3)	sst	EP			
Pr 142	S	19.2 h	6.36 (+4)	1.9 (-3)	sst	EP			
Pr 142m	S	14.6 m	1.71 (+4)	5.1 (-4)	sst	EP			
Pr 143	S	13.57 d	1.49 (+6)	4.5 (-2)	sst	EP			
Pr 144m	S	7.2 m	1.44 (+4)	4.3 (-4)	sst	EP			

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Pr 145	S	5.98 h	1.02 (+6)	3.1 (-2)	sst	EP			
Pr 146	S	24 m	8.18 (+5)	2.5 (-2)	sst	EP			
Pr 147	S	12 m	6.48 (+5)	1.9 (-2)	sst	EP			
Pr 148	S	1.98 m	5.13 (+5)	1.5 (-2)	sst	EP			
Pr 149	S	28 s	3.59 (+5)	1.1 (-2)	sst	EP			
Pr 150	S	10 s	2.39 (+5)	7.2 (-3)	sst	EP			
Nd 144	S	2.1(+15)a	1.21 (-9)	3.6 (-17)	sst	EP			
Nd 147	S	10.98 d	6.60 (+5)	2.0 (-2)	sst	EP			
Nd 149	S	1.73 h	3.80 (+5)	1.1 (-2)	sst	EP			
Nd 151	S	12.4 m	2.01 (+5)	6.0 (-3)	sst	EP			
Nd 152	S	11.4 m	1.38 (+5)	4.1 (-3)	sst	EP			
Pm 147	S	2.62 a	1.29 (+5)	3.9 (-3)	sst				
Pm 148	S	5.37 d	2.91 (+5)	8.7 (-3)	sst				
Pm 149m	S	41.3 d	6.46 (+4)	1.9 (-3)	sst				
Pm 149	S	53.1 h	5.90 (+5)	1.8 (-2)	sst				
Pm 150	S	2.7 h	1.24 (+3)	3.7 (-5)	sst				
Pm 151	S	28 h	2.02 (+5)	6.1 (-3)	sst				
Pm 152	S	15 m	1.42 (+5)	4.3 (-3)	sst				
Pm 152m	S	7.5 m	3.29 (+3)	9.9 (-5)	sst				
Pm 153	S	5.3 m	9.05 (+4)	2.7 (-3)	sst				
Pm 154	S	1.6 m	5.03 (+4)	1.5 (-3)	sst				
Pm 154m	S	2.6 m	1.04 (+4)	3.1 (-4)	sst				
Sm 147	S	106 (+9)a	1.33 (-6)	4.0 (-14)	sst	EP			
Sm 148	S	7 (+15)a	5.91(-11)	1.8 (-18)	sst	E			
Sm 151	S	93 a	2.40 (+?)	7.2 (-6)	sst	E			

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classifi-cation	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Sm 155	S	22.4 m	3.87 (+4)	1.2 (-3)	sst	E			
Sm 156	S	9.4 h	2.43 (+4)	7.3 (-4)	sst	E			
Sm 157	S	8 m	1.48 (+4)	4.4 (-4)	sst	E			
Eu 152	S	12.4 a	4.18	1.3 (-7)	sst	EP			
Eu 154	S	8.5 a	1.70 (+4)	5.1 (-4)	sst	EP			
Eu 155	S	4.96 a	9.94 (+3)	3.0 (-4)	sst	EP			
Eu 156	S	15.2 d	2.48 (+5)	7.4 (-3)	sst	EP			
Eu 157	S	15.15 h	1.53 (+4)	4.6 (-4)	sst	EP			
Eu 158	S	46 m	8.99 (+3)	2.7 (-4)	sst	EP			
Eu 159	S	18.7 m	4.08 (+3)	1.2 (-4)	sst	EP			
Gd 152	S	110(+12)a	2.01 (-13)	6.0 (-21)	sst	EP			
Gd 153	S	241.6 d	5.00 (+1)	1.5 (-6)	sst	EP			
Gd 159	S	18.56 h	5.38 (+3)	1.6 (-4)	sst	EP			
Tb 160	S	72.1 d	1.01 (+3)	3.0 (-5)	sst	EP			
Tb 161	S	6.9 d	1.09 (+3)	3.3 (-5)	sst	EP			
Dy 165	S	2.35 h	4.89 (+2)	1.5 (-5)	sst	EP			
Dy 166	S	81.5 h	5.45	1.6 (-7)	sst	EP			
Ho 166	S	26.7 h	1.10 (+2)	3.3 (-6)	sst	EP			
Ho 166m	S	1 200 a	1.48 (-3)	4.4 (-11)	sst	EP			
Ta 182	A	115 d	1.28 (+4)	3.8 (-4)	sst	EP			
W 185	A	75.1 d	8.56	3.5 (-5)	m	EP			
Th 230	AS	77 000 a	3.76 (-6)	3.8 (-14)	a				
Th 232	A	14.05(+9)a	2.58 (-11)	2.6 (-19)	a				
Pa 231	AS	32 500 a	6	6.0 (-8)	a				
Pa 233	AS	27 d	3.34 (-1)	3.3 (-9)	a				

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classification	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
U 232	AS	71.7 a	1.01 (-3)	1.0 (-11)	a				
U 233	AS	0.159(+6)a	2.46 (-5)	2.5 (-13)	a				
U 234	AS	0.244(+6)a	1.68 (-1)	1.7 (-9)	a				
U 235	AS	0.704(+9)a	1.85 (-2)	1.9 (-10)	a				
U 236	AS	23.4(+6)a	3.10 (-1)	3.1 (-9)	a				
U 237	AS	6.75 d	1.98 (+6)	1.1 (-2)	a				
U 238	AS	4.47(+9)a	3.17 (-1)	3.2 (-9)	a				
U 239	AS	23.5 m	2.20 (+7)	2.2 (-1)	a				
Np 236	AS	1.29(+6)a	2.00 (-5)	2.0 (-13)	a				
Np 237	AS	2.14(+6)a	3.45 (-1)	3.5 (-9)	a				
Np 238	AS	50.8 h	4.89 (+5)	4.9 (-3)	a				
Np 239	AS	2.355 d	2.19 (+7)	2.2 (-1)	a				
Np 240	AS	65 m	4.63 (+4)	4.6 (-4)	a				
Pu 237	AS	45.6 d	3.63	3.6 (-8)	a				
Pu 238	AS	87.75 a	2.81 (+3)	2.8 (-5)	a				
Pu 239	AS	24 400 a	3.21 (+2)	3.2 (-6)	a				
Pu 240	AS	6 540 a	4.91 (+2)	4.9 (-6)	a				
Pu 241	AS	14.89 a	1.06 (+5)	1.1 (-3)	a				
Pu 242	AS	0.387(+6)a	1.32	1.3 (-8)	a				
Pu 243	AS	4.96 h	4.15 (+5)	4.2 (-3)	a				
Am 241	AS	433 a	7.62 (+1)	7.6 (-7)	a				
Am 242m	AS	152 a	8.25	8.3 (-8)					
Am 242	AS	16 h	6.63 (+4)	6.6 (-4)					
Am 243	AS	7 400 a	1.81 (+1)	1.8 (-7)	a				
Am 244	AS	10.1 h	1.40 (+5)	1.4 (-3)	a				

5 (-4) = 5×10^{-4}.

TABLE 2.3.1.2.1. IMPORTANT NUCLIDES IN LWRS.

Nuclide	A/S	Half-life	Core inventory in Ci/t of heavy metal	Primary loop activity (pl) in Ci/t of water	pl classifi-cation	Essential trace element	Spiking factor (SP) (SP avgd.)	Total time of SP (Avg. SP time)	Primary loop activities of the KWU [8] with 10% clean-up of the primary loop without degasification
Cm 242	AS	163 d	3.38 (+4)	3.4 (-4)	a				
Cm 243	AS	30 a	4.22	4.2 (-8)	a				
Cm 244	AS	18.099 a	2.32 (+3)	2.3 (-5)	a				
Cm 245	AS	8 532 a	3.14 (-1)	3.1 (-9)	a				
Cm 246	AS	4 820 a	6.35 (-2)	6.3 (-10)	a				

5 (-4) = 5×10^{-4}.

Literature Table

No.	Nuclide	Cond. - Lit.	Spike-Lit.
1	Cr 51	17(8,19,20,21,22, 23,25,26)	22(17)
2	Mn 54	22(8,17,19,20,23, 26,27)	17(22)
3	Mn 56	23	20
4	Fe 59	22(8,19,20,21,23,27)	20
5	Co 58	15(8,13,17,19,20,21, 22,23,26,27)	17(20,22)
6	Co 60	23(8,13,15,19,20,21, 22,,26,27)	17(22)
7	In 55	21	-
8	Kr 85m	26(17,19,25)	17
9	Kr 87	26(8,19,27)	-
10	Kr 88	26(8,19,27)	-
11	Zr 95	20(8)	-
12	Zr 97	20	-
13	Sb 122	22	22
14	Sb 124	22(17,23)	17
15	I 131	14(8,13,17,19,22,24, 25,27)	17(13,14,19,29)
16	I 133	26(8,17,19,22,25,27)	26(17,25)
17	Xe 133	14(8,17,19,22,24,26, 27)	17(14)
18	Xe 135	26(8,19,22,27)	14
19	Cs 134	14(8)	14
20	Cs 137	14(8,25,30)	30 (14)
21	W 185	20	-

Legend to Literature Table

No. = Number.

Nuclide = Element - Mass number.

Con. lit. = Literature on primary loop concentrations

-- Number in front of parentheses: Literature citation for the measured primary loop activity (column 5 of the previous table).

-- Number(s) in the parentheses: Literature citation(s) which was (were) analyzed regarding the special nuclide activity.

-- Underlined numbers: Literature citations which contain only calculated values.

Spike lit. = Literature on the spiking factors

-- Number in front of parentheses: Literature citation on measured primary loop activity variation from which the spiking factor was obtained (column 8 in the table).

-- Number(s) in the parentheses: Literature citation(s) which were also evaluated with regard to the spiking factors or spiking time.

2.3.2. Important Nuclides of the KWU and GRS - A Comparison

Table 2.3.2.1 summarizes the nuclides which were selected by the KWU and the GRS as being of importance. This selection is decisive for subsequent activity calculations. Nuclides which are omitted from this selection can no longer play any role during subsequent emission and immission calculations performed on this basis.

The summary shows how arbitrarily the selection of the nuclides was made. This must be considered from the standpoint that the KWU and the GRS selected only 19 and 28 or 54, respectively, of the more than 500 nuclides which are generated in a NPP and then maintain that their considerations are basically "conservative".

Unfortunately, we cannot provide the corresponding table from the safety report for the Biblis B NPP since the Board of Directors of the RWE refused to give out the safety reports. This involves a matter of company security.

TABLE 2.3.2.1. IMPORTANT NUCLIDES OF THE KWU AND GRS.

Grohnde Safety report 1973 (7)	Neckarwestheim safety report 1980 (8)	German Risk Study most important nuclides 1980 (5)
Kr 85m Kr 85 Kr 87 Kr 88 Xe 133 Xe 135	Kr 85m Kr 85 Kr 87 Kr 88 Xe 131m Xe 133m Xe 133 Xe 135m Xe 135 Xe 138	Kr 85m Kr 85 Kr 87 Kr 88 Xe 133 Xe 135
J 131 J 132 J 133 J 134 J 135	J 129 J 131 J 132 J 133 J 134 J 135	J 131 J 132 J 133 J 134 J 135
Sr 90 Cs 137	H 3 Rb 88 Sr 90 Cs 134 Cs 137 Cs 138	Rb 86 Sr 89 Sr 90 Sr 91 Cs 134 Cs 136 Cs 137
Cr 51 Mn 54 Fe 59 Co 58 Co 60 Zr 95	Cr 51 Mn 54 Fe 59 Co 58 Co 60 Zr 95	Co 58 Co 60 Y 91 Zr 95 Nb 95 Zr 97 Mo 99 Tc 99m Ru 103 Ru 105 Rh 105 Ru 106 Sb 127 Te 127m Te 127 Sb 129 Te 129m Te 129 Te 131m Te 132 Ba 140 La 140 Ce 141 Ce 143 Pr 143 Ce 144 Nd 147 Pu 238 Np 239 Pu 239 Pu 240 Pu 241 Am 241 Cm 242 Cm 244

[J = I]

2.3.3. Essential and Potentially Essential Trace Elements Which were not Included in the "Important Nuclides" of the KWU and the GRS

In biochemistry metals have long been known as "essential trace elements". In recent years analysis methods have made it possible to recognize many previously unconsidered elements, primarily metals, as essential (compare Figure 6 from [1]).

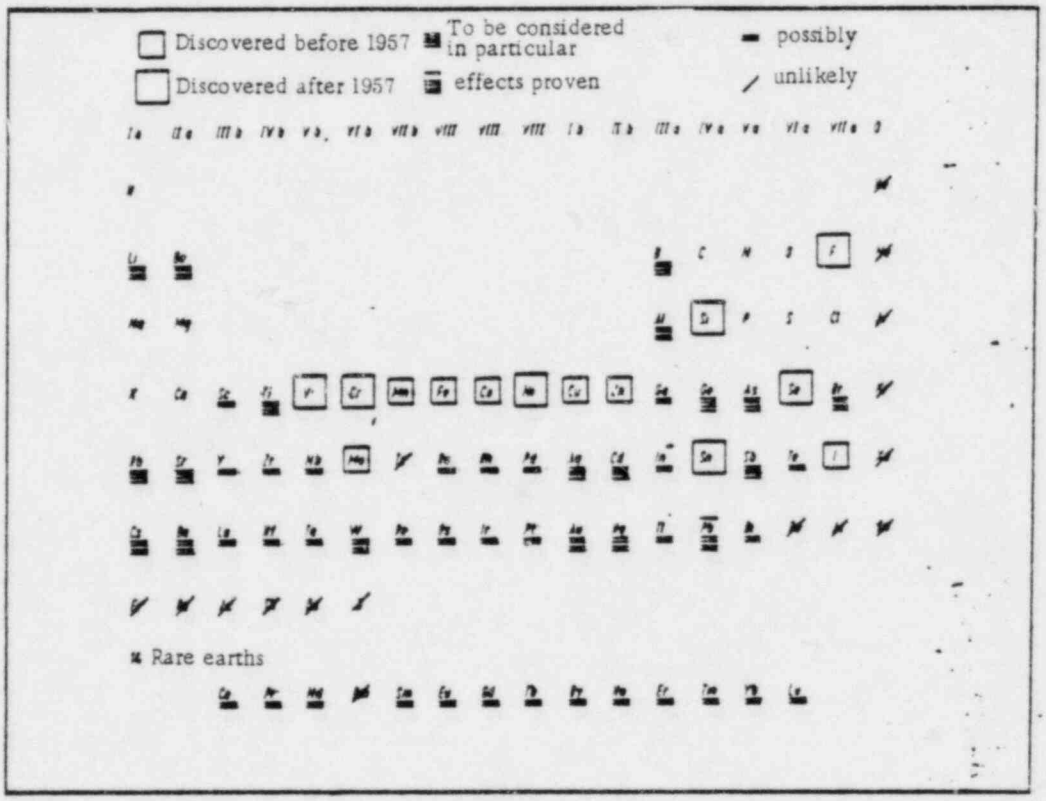


Figure 2.3.3.1. Distribution in the Periodic Table of the Trace Elements Which are Necessary for Warm-Blooded Animals.

Essential elements are indispensable for the metabolism of living organisms. Both an excess and a deficit lead to disruptions in metabolism and ultimately to diseases. Although metals comprise only approximately 3% of the human body, life is much more dependent on them than indicated by this figure [2].

The special role of the essential trace elements also affects the importance of the individual nuclides which are obtained from nuclear facilities. The work of Bruland et al. [3] shows this, by way of example, for the isotopes of cobalt (Co). From this new standpoint the isotopes of Co were decisively attributed a much greater importance in the assessment of radiation damage

due to emissions from nuclear plants; this is due in this case primarily to the longer biological half-life which cobalt has in the form of the vitamin B-12 complex.

The state of the art has continued to develop. The development, distribution, emission and induced damage of essential and potentially essential trace elements must be given particular attention in the case of nuclear facilities. Whether from this new standpoint it is even defensible to expect model emission mixtures during normal operation must be strongly questioned. In the event of incidents and accidents, the 54 nuclides selected in the German Risk Study [5] as particularly relevant do not correspond to the state of the art.

Table 2.3.3.2 summarizes the essential and potentially essential nuclides (compare [1] Figure 6) which are not included either in the cited safety reports of the KWU or in the 54 nuclides of the German Risk Study (Technical Volume 6, Table 6-3-1).

TABLE 2.3.3.2. ESSENTIAL AND POTENTIALLY ESSENTIAL ISOTOPES OF TRACE ELEMENTS•WHICH ARE NOT INCLUDED EITHER IN THE SAFETY REPORTS OF THE KWU [7], [8], OR IN THE GERMAN RISK STUDY [5].

Isotope	Symbol	Isotope	Symbol	Isotope	Symbol	Isotope	Symbol
Be 10	EP	Se 83	E	Nb 97m	EP	Pd 113	EP
Al 28	EP	Se 83m	E	Nb 98	EP	Pd 114	EP
Si 31	E	Se 84	E	Nb 101	EP	Ag 108	EP
Sc 46	EP	Se 85	E	Nb 102	EP	Ag 109m	EP
Sc 47	EP	Se 86	E	Mo 93	E	Ag 110	EP
Sc 48	EP	Se 87	E	Mo 101	E	Ag 110m	EP
V 52	E	Se 88	E	Mo 102	E	Ag 111	EP
Cr 55	E	Br 82	EP	Mo 103	E	Ag 111m	EP
Mn 56	E	Br 82m	EP	Mo 104	E	Ag 112	EP
Fe 55	E	Br 84m	EP	Mo 105	E	Ag 113	EP
Co 60m	E	Br 85	EP	Mo 106	E	Ag 115	EP
Co 61	E	Br 86	EP	Ru 107	EP	Cd 113	EP
Ni 59	E	Br 86m	EP	Ru 108	EP	Cd 115	EP
Ni 63	E	Br 87	EP	Ru 109	EP	Cd 115m	EP
Ni 65	E	Br 88	EP	Ru 110	EP	Cd 117	EP
Cu 64	E	Br 89	EP	Rh 103m	EP	Cd 117m	EP
Zn 65	E	Br 90	EP	Rh 104	EP	Cd 118	EP
Zn 72	E	Rb 86	EP	Rh 105m	EP	In 114m	EP
Ga 72	EP	Rb 87	EP	Rh 106	EP	In 115	EP
Ga 73	EP	Zr 93	EP	Rh 107	EP	In 115m	EP
Ge 75	EP	Zr 98	EP	Rh 109	EP	In 116	EP
Ge 77	EP	Zr 99	EP	Rh 109m	EP	In 117	EP
Ge 78	EP	Zr 100	EP	Rh 110	EP	In 117m	EP
Ge 79	EP	Zr 101	EP	Rh 111	EP	In 119m	EP
Ge 80	EP	Zr 102	EP	Pd 107	EP	Sn 117m	E
Ge 81	EP	Nb 93m	EP	Pd 109	EP	Sn 119m	E
Ge 82	EP	Nb 94	EP	Pd 109m	EP	Sn 121	E
Se 79	E	Nb 95m	EP	Pd 111	EP	Sn 121m	E
Se 81	E	Nb 96	EP	Pd 111m	EP	Sn 123	E
Se 81m	E	Nb 97	EP	Pd 112	EP	Sn 123m	E

For all of the nuclides listed in the table (for instance, Sn 117m, Sn 119m, Sn 121, Sn 123, Sb 126), no inhalation or ingestion factors are listed in the "General Calculation Fundamentals" [4].

TABLE 2.3.3.2. CONTINUED

Isotope	Symbol	Isotope	Symbol	Isotope	Symbol	Isotope	Symbol
Sn 125	E	J 138	E	Pr 142m	EP	Dy 165	EP
Sn 125m	E	Cs 134m	EP	Pr 144m	EP	Dy 166	EP
Sn 127	E	Cs 135	EP	Pr 145	EP	Ho 166	EP
Sn 127m	E	Cs 135m	EP	Pr 146	EP	Ho 166m	EP
Sn 128	E	Cs 138	EP	Pr 147	EP	Ta 182	EP
Sn 130	E	Cs 138m	EP	Pr 148	EP	W 185	EP
Sn 131	E	Cs 139	EP	Pr 149	EP		
Sn 132	E	Cs 140	EP	Pr 150	EP		
Sb 122	EP	Cs 141	EP	Nd 144	EP		
Sb 124	EP	Cs 142	EP	Nd 149	EP		
Sb 125	EP	Ba 135m	EP	Nd 151	EP		
Sb 126	EP	Ba 137m	EP	Nd 152	EP		
Sb 131	EP	Ba 139	EP	Sn 147	E		
Sb 133	EP	Ba 141	EP	Sn 148	E		
Te 123	EP	Ba 142	EP	Sn 151	E		
Te 123m	EP	Ba 143	EP	Sn 155	E		
Te 125m	EP	Ba 144	EP	Sn 156	E		
Te 131	EP	Ba 145	EP	Sn 157	E		
Te 133	EP	La 138	EP	Eu 152	EP		
Te 133m	EP	La 141	EP	Eu 154	EP		
Te 134	EP	La 142	EP	Eu 155	EP		
Te 135	EP	La 143	EP	Eu 156	EP		
Te 137	EP	La 145	EP	Eu 157	EP		
J = I J 128	E	La 146	EP	Eu 158	EP		
J 129	E	La 147	EP	Eu 159	EP		
J 130	E	Ce 145	EP	Gd 152	EP		
J 130m	E	Ce 146	EP	Gd 153	EP		
J 134m	E	Ce 147	EP	Gd 159	EP		
J 136	E	Ce 149	EP	Tb 160	EP		
J 137	E	Pr 142	EP	Tb 161	EP		

For all of the nuclides listed in the table (for instance, Sn 117m, Sn 119m, Sn 121, Sn 123, Sb 126), no inhalation or ingestion factors are listed in the "General Calculation Fundamentals" [4].

2.3.4. Determination of the Bandwidths for the Parameter Study

In the parameter study both the half-lives and the primary loop activities of standard nuclides are varied. Table 2.3.4.1 lists several nuclides which are comparable to the standard nuclides.

The primary loop concentrations are varied from 10^{-8} Ci/t to 10^{2} Ci/t. The lower boundary of 10^{-8} Ci/t corresonds to the lowest nuclide-specific detection limit which is required in KTA Rule 1504 [28]. The upper limit of 10^{2} Ci/t approximately corresonds to the design value of I-131 multiplied by a spiking factor of 100.

TABLE 2.3.4.1. "STANDARD NUCLIDES" USED IN THE PARAMETER STUDY AND SELECTED COMPARABLE NUCLIDES OF THE CORE INVENTORY

Decay constants of the standard nuclide	Half-lives T 1/2 of the standard nuclides (s, h, d, a)	Comparable nuclides	
		Nuclide	T 1/2
10^{-12}	22,000 a	Pu 239	24,390 a
		Pa 231	32,500 a
10^{-10}	220 a	Am 242m	152 a
		Sn 121m	50 a
		Sn 151	93 a
10^{-8}	2.2 a	Cs 134	2.06 a
		Sb 125	2.77 a
		Eu 155	4.96 a
10^{-6}	8.06 d	I 131	8.04 d
		Sb 126	12.40 d
		Xe 133	5.29 d
10^{-4}	1.92 h	Kr 87	1.27 h
		As 78	1.50 h
10^{-2}	69 s	Ge 79	42 s
		Br 86	54 s
		As 81	34 s
		Se 85	33 s
10^{0}	0.69 s	Sr 98	0.60 s
		Y 98	1 s

a = Year

d = Days

h = Hours

s = Seconds

2.3.5. Bibliography for 2.3

1. Betke, Klaus and Frank Bidlingmaier (publisher): "Trace Elements in the Development of Man and Animals." "Ignored Elements in Infant Feeding". Symposium in Rottach-Egern, 1974, Munich-Berlin-Vienna, 1975.

2. Williams, D. R.: "Metals, Ligands and Cancer", CHEMICAL REVIEW, Vol. 72, p. 203, 1972.

3. Bruland, W., B. Franke and D. Tuefel: "Transfer of Organically Bound Radionuclides Through Food Chains to Man: Model-Example with Radio-Cobalt and Vitamin B_{12}". International Symposium of Biological Implications of Radionuclides Released from Nuclear Industries, IAEA-SM-237/17, Vienna, 3/26-30/79.

4. Federal Ministry of the Interior (BMI), Bulletin: Bulletin: "General Calculation Fundamentals for the Radiation Exposure of Radioactive Discharges with Exhaust Air or in Surface Bodies of Water." Guideline for § 45 of the Radiologic Protection Ordinance, RS II-515603/2, 8/15/79.

5. Federal Minister for Research and Technology (publisher): "Deutsche Risikostudie Kernkraftwerke. Fachband 6: Ermittlung der Spaltproduktfreisetzung" [German Risk Studies on NPPs. Technical Volume No. 6: Determination of Fission Product Release]. Table F 6-3-1, p. 11, Cologne-Bonn, 1979.

6. Letter from the IFEU - Institute for Energy and Environmental Research to the KWU dated 9/18/79 and response from the KWU dated 10/2/79.

7. The Power Plant Union (KWU) (publisher): "Sicherheitsbericht zum Kernkraftwerk Grohnde" [Safety Report on the Grohnde NPP], Erlangen, 1973.

8. The Power Plant Union (KWU) (publisher): "Sicherheitsbericht zum Kernkraftwerk Neckarwestheim" [Safety Report on the Neckarwestheim NPP], Erlangen, 1980.

9. Seelmann-Eggebert, W., G. Pfenning and H. Muenzel: "Nuklidkarte" [Nuclide Chart], Karlsruhe Nuclear Research Center, 1974.

10. Bell, N. J., 'ORIGEN - The ORNL Isotope Generation and Depletion Code", (ORNL-4628, UC Mathematics and Computers), 1973.

11. Baden Technical Supervisory Association (publisher): "Activity Release in the Case of Accidents in NPPs with LWRs via the Exhaust Air Pathways Within the Framework of a Guideline for § 28, paragraph 3, of the Radiologic Protection Ordinance (Status 12/14/79), 1979.

12. Federal Health Office and Labor Association for Environmental Matters (publisher): "Radioecology Symposium, 10/15-16/81, Stuttgart, 1981.

13. Roofthooft: "Chemische Betriebserfahrungen in Doel and Tihagne" [Chemical Operating Experience in Doel and Tihagne], 1978.

14. Neeb, K. H. and R. Korn: "Examination and Accumulation of Operating Experience for the Optimization of Water Chemistry", BMFT (publisher) - Research Report K 73-12, Leopoldshafen, 1973.

15. Kirchenmayer: "Activated Corrosion Products in LWR Loops", KWU (publisher), BMFT-RS 209, "First Technical Report", 1978.

16. Kausz, I.: "LWR mit Tritiumentzugsanlagen" [LWRs with Tritium Removal Systems], 1973.

17. Schroeder, H. J., A. Dorr and G. Paffrah: "Chemistry in the Biblis NPP, Operating Experience", VGB Conference, "Chemistry in the Power Plant", 1979.

18. Radiologic and Environmental Research Company (GSF) (publisher): "Nachweis von α-Strahlern in Abwasser, Primaerkreis und Abluft" [Detection of α-Emitters in Waste Water, Primary Loop and Exhaust Air], Munich.

19. Kausz, I.: "Procedures for Reducing Radioactive Discharges", KRAFTWERK UND UMWELT, pp. 162-168, 1977.

20. Kennedy, A. J.: "Corrosion Studies in the Primary System at Oconee I", Babcock and Wilcox.

21. Reinhard, K. H.: "Gases and Corrosion Products in the Primary Loop of the NS [Nuclear Ship] Otto Hahn", VGB Conference "Chemistry in the Power Plant", pp. 28-36, 1976.

22. Wille, H.: "Plant Engineering Experience with Primary Loop Clean-Up Systems in NPPs with PWRs and BWRs", VGB KRAFTWERKSTECHNIK, Vol. 57, No. 10, October 1977.

23. Paffrath, G. and R. Ambros: "Activation Products and Dose Rates Using the Example of the Biblis NPP, Unit A", VGB KRAFTWERKSTECHNIK, Vol. 2, February 1980.

24. Annual Report of the ABE [Committee for the Exchange of Operating Experience], in Working Group I "Engineering and Industry" of the German Atomic Forum: "Operating Experience with NPPs in the FRG", in: ATOMSTROM, 1977.

25. Neeb, K. H.: "Origin and General Behavior of Increased Release of Fission Iodine from Defective Fuel Rods Upon Shutdown of Pressurized Water Reactors", SIEMENS FORSCHUNGS- UND ENWICKLUNGSBERICHT, Vol. 8, No. 2, 1979.

26. Brunner, R: "Chemical Operating Experience in the Stade NPP During the First Three Operating Cycles", in: VGB KRAFTWERKSTECHNIK, 56th year, Vol. 2, February 1976.

27. Caruthers, G. F.: "Radioiodine Behavior in the Reactor Coolant System During Transient Operation", CENPD - 180, March 1976 and March 1977.

28. Safety Rule of the KTA 1504, Version of 6/78: "Measurement of Liquid Radioactive Substances to Monitor Radioactive Discharges".

29. Chubb, W., E. J. Piplica and R. J. Lutz: "The Iodine Spiking Phenomenon in Pressurized Water Reactor Coolants", (W-NES).

30. Schenk, H. and A. Mayr: "The Obrigheim NPP: Experience in the First Year of Operation", ATOMWIRTSCHAFT (ATW), Vol. 7, p. 324, 1970.

2.4 SG Heating Tube Leaks

- Introduction

The SGs separate the radioactively contaminated primary coolant loop from the water/steam loop which drives the turbines.

Each of the four SGs of a KWU-1300 MW_{el} PWR contains U-shaped heat exchanger tubes with a total length of approximately 65 km and a wall thickness of approximately 1.2 mm.

It is immediately obvious that during the operation of a NPP there may be more or less large leaks in the total of 16,240 SG heating tubes with a total length of approximately 260 km. Both mechanical and chemical factors, as well as human error, can lead to such leaks. Leaks are therefore initially independent of the materials being used at the time. Both in the case of the material Inconel 600 which is used and so highly praised by Westinghouse and in the case of the material Incoloy 800 which is also praised by the KWU and is used in all NPPs except for Obrigheim, leaks occur over the years. As examples of KWU steam generators, let us mention the leaks in Biblis B, Stade and Borselle.

In contrast to these considerations, the KWU philosophy has to date virtually completely prevailed in the FRG; it says:

> The material used by the KWU, Incoloy 800, and the design of the SGs is so much better than the material Inconel 600 used by other countries (for instance, U.S.A.) and their SG designs that SG leaks are not to be expected in the FRG.

As documentation, let us cite an excerpt from the Administrative Law Case on the Wyhl NPP dated 11/13/80 [14]. It is stated that secondary loop emissions do not need to be taken into account since the SGs have exhibited no further leaks since the use of the material Incoloy 800 began. In particular, it is stated:

> "The defendant [this word is uncertain] (The Sued NPP, Incorporated, author's note) has represented (by KWU experts, author's note) that since SG heating tubes made of Incoloy 800 have been used, there have been no more SG leaks and consequently the secondary loops of the Biblis A and B, Neckarwestheim, Borselle and Atucha Plants have been free of activity."

Note: At the time of this statement in court at the end of 1980, the KWU was quite well aware of the SG leaks at Borselle and Biblis.

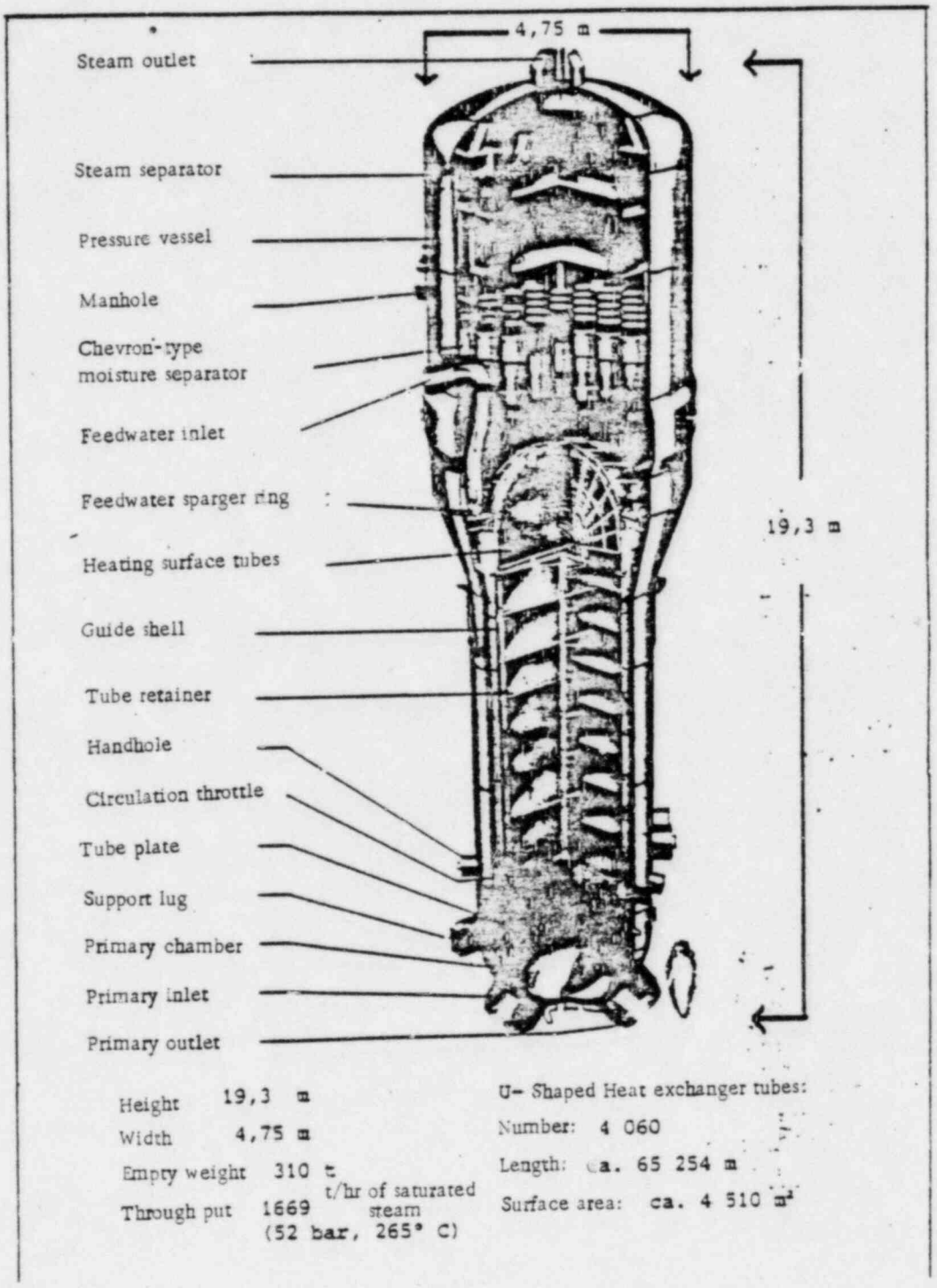


Figure 2.4.1. Steam Generator for a PWR (Biblis, Engineering Babcock and Wilcox).

This safety philosophy of an activity-free secondary loop system of the KWU has implications for the technical design of the secondary loops of German NPPs:

-- During power operation, steam from the secondary system is used to humidify the air of the switchgear plant building (control room, computer center, etc.). In the case of SG leaks, the radioactivity contained in the steam will thus be uniformly distributed through the air conditioning system. Imagine the catastrophic effects in the case of a partial core melt-down caused by a LOCA due to a SG heating tube rupture.

-- This direct humidification of the air by secondary loop water was verified for the Biblis A, B and Grohnde NPPs [15, [16].

-- Steam is continuously released directly to the outside from the power house. In the case of SG heating tube leaks, this will lead to unfiltered, unmeasured and thus unmonitored emissions.

In the next section past leaks and their causes are summarized. It is shown that the reactor-engineering limitation of SG heating tube leaks by the N-16 scram [system] contradicts the philosophy of the reactor protection system. In addition, the bandwidth [range] of SG heating tube leaks for the parameter study is established.

2.4.1. Previous SG Heating Tube Leaks

An evaluation by Stevens-Guille [17] of all SG leaks up to 1971 shows that the mean time between two instances of SG damage is approximately 0.6 year. It is to be assumed that this mean time between SG damage will decrease as the age of the plant increases.

The following table summarizes the SG leaks and their causes up to the year 1975 as noted worldwide in the summary works of the GRS, etc. [1-9, 11], and for the FRG from other sources [10], [12], [13], up to September 1981. Since both the obligation to report such damage and the publication policy of the operators and/or the supervisory authorities are handled in different ways internationally and, with two exceptions, Table 2.4.1.1 contains only published SG damage cases, these data are representative but are certainly not reliable in terms of absolute numbers. The causes of the SG damage are generally not completely proven so that some of the data are based solely on assumptions.

Reactor plant	Type (MW)	Year of start-up	No. of SGs heating tubes per SG)	No. of heating tubes tested	No. of heating tube damages	No. of leaky heating tubes among them	Sum of heating tube damages (cumula.)	Radiation burden of amount of leakage	Cause of defect, if given	Bibl.
Besnau I	PWR 364	1969	2 SG (2604)	1972:60%	1971:400 1972:500 1973:8		1973:960	1973:6-9 R/hr within SG	1971:Crack area above tube sheet. 1972:Defect in area directly above tube sheet. 1973:Abrasion fretting corrosion in clearly outlined patterns in the hot leg mud deposits up to 120 mm high on the tube sheet.	[1]
				1974:27%	1974:2 1975:5		1975:982		1974:One instance of damage inside of mud zone and one instance of damage outside of mud zone. 1975:Uniform erosion and intercrystalline corrosion in the area of the mud deposits and on the tube sheet.	[2] [3]
Besnau II Inconel 600	PWR 364	1971	2 SG (2604)	1974:93%	1973: 0 1974:39 1975:56	1975:1	1973:0 1974:19 1975:108		1974:Uniform erosion in mud deposits, tube sheet. 1975: "	[2] [3]
Biblis A Incoloy	PWR 1204	1974	4 SG (4060)	1974:9%	1974:0 1975:0					
Biblis B Incoloy	PWR 1300	1977	4 SG		1979:1	1979:1			Nov. 1979:Leak due to pitting corrosion and wastage.	[10]
Borselle Incoloy	PWR 477	1973			1980:40	1980:<1			Reduction of wall thickness of more than 50%.	[11]

Figure [sic, should be "Table"] 2.4.1.1. Overview of SG Heating Tube Leaks Throughout the World.

Reactor plant	Type (MW)	Year of start-up	No. of SGs heating tubes per SG)	No. of heating tubes tested	No. of heating tube damages	No. of leaky heating tubes among them	Sum of heating tube damages (cumula.)	Radiation burden of amount of leakage	Cause of defect, if given	Bibl.
R.E. Ginna Inconel 600	PWR 515	1969	2 SG (3260)	1973:65%	1973:0 1974:22 1975:60 1976:41	1975:2	1974:22 1975:82 1976:123		1974:Erosion close to the tube sheet. 1975:Uniform erosion and intercrystalline corrosion, tube sheet area. Some damage due to mud residues.	[2] [3]
Haddam Neck Inconel 600	PWR 575		4 SG (3794)	1973:30%	1972:14 1973:10		1973:26 1974:26		1972:Flaw in area directly above tube sheet. 1973:Eight damage points in the center of the hot leg above the tube sheet, two in the tube bend area, deposits on the tube sheet.	[1]
Jose Cabrara Inconel 600	PWR 160	1968	1 SG (2604)	1974:3%	1974:3		1974:3		No data.	[2]
KWU Stade Incoloy	PWR 662	1972	4 SG (3000)		1973:0 1974:0 1975:0 1981:1	1981:1		1981:160 l	1975:Two tubes pulled for inspection. 1981:Corrosion of a tube by wastage, reactor shutdown.	[3] [12]
KKW Gundremmingen	BWR 337	1966	3 SG (1929)		1974:14 1975:119		1974:14 1975:133		1974:Defects in bundle center near partition. 1975:Corrosion in center of tube sheet.	[2] [3]
KKW Litgan	BWR 256	1968	2 SG (5000)		1975:0		1974:20 1975:20		No data.	[2] [3]

Figure [sic, should be "Table"] 2.4.1.1. Overview of SG Heating Tube Leaks Throughout the World. [Continuation]

Reactor plant	Type (MW)	Year of start up	No. of SGs heating tubes per SG)	No. of heating tubes tested	No. of heating tube damages	No. of leaky heating tubes among them	Sum of heating tube damages (cumula.)	Radiation burden of amount of leakage	Cause of defect, if given	Bibl.
KKW Obrigheim Inconel 600	PWR 345	1968	2 SG (2605)	1972:992 1973:1164 1974:1254 1975:1120	1972:98 1973:61 1974:60 1975:41	1971:6 1972:29 1973:3 1974:1 1975:5	Jan.81: 3 t/hr in SG II Conse-quence: 10 Ci Xe, 700 mCi I-131, 3 mCi Co-58 in the tank water of the SG 1979: approx. 200 l/day 600 l/day		Leaks in the first year of operation in SG I (approx. 1 l/ /day). 1971:Leak, cause unknown. After 1971: Increase in blowdown from 10 t/day to 200 t/day. 1972:44 instances of damage in the area directly above tube sheet, 17 instances of damage in the elbow area, 37 instances of damage, crack area above tube sheet. 1973:Intergranular stress corrosion cracks above the tube sheet in the bundle center. 1974:Stress corrosion cracking, SG I, bundle center near tube sheet, SG II, innermost tube sheet in bend area. 1979:At beginning of operating time. At end of operating time.	[4] [1] [2] [11]
Mihame I Inconel 600	PWR 340	1970	2 SG (4426)	1974:200	1973:2011 1974:5 1975:12		1974:2127 1975:2139		1973:Thinning of walls due to local thermal hydraulic processes in the area of the tube bends under the oscillation damping belts. 1975:Uniform erosion in the area of the tube bend support.	[1] [3]

Figure [sic, should be "Table"] 2.4.1.1. Overview of SG Heating Tube Leaks Throughout the World. [Continuation]

Reactor plant	Type (MW)	Year of start up	No. of SGs heating tubes per SG)	No. of heating tubes tested	No. of heating tube damages	No. of leaky heating tubes among them	Sum of heating tube damages (cumula.)	Radiation burden of amount of leakag-	Cause of defect, if given	Bibl.
Mihame II Inconel 600	PWR 500	1972	2 SG (3260)		1973:0 1974:0 1975:266		1973:0 1974:0 1975:266			[3]
Indian Point RS Material	PWR 265		4 SG (811)		1973:153 1974:16		1973:249 1974:265	1973: During repairs on 6 tubes a radiation 6-15 R/hr for a total of 3500 man-rem.	1973:Suspicion of stress corrosion in the area of the tube bend supports. . • burden of arose,	[1] [2]
Indian Point 2	PWR 920	1973	4 SG (3260)		1973:0 1974:0 1975:3		1973:0 1974:0 1975:5			[1] [2] [3]
Oconee I Inconel 600	PWR 922	1973	2 SG (15531)	1974:3s	1974:2 1975:0 1976:1 Apr 78:2 Oct 78:7		1974:2 1975:2 1976:3	1976, 3.0 l/min of primary water Leakage reached boundary value (value not given)	1976:180° circumferential crack on a tube due to component flaw. April 78:Cracks in two tubes. Oct. 78:Human failure: 7 tubes with reduced walls (40% reduction) not closed; rather 7 intact tubes were closed due to a counting error.	[2] [3] [5] [6] [7]
Oconee II Inconel 600	PWR 922	1973	2 SG (15531)		1975:0		1975:0			[3]
Oconee III Inconel 600	PWR 92 [sic]	1974	2 SG (15531)				1975:0			[3]

Figure [sic, should be "Table"] 2.4.1.1. Overview of SG Heating Tube Leaks Throughout the World. [Continuation]

Reactor plant	Type (MW)	Year of start up	No. of SGs heating tubes per SG)	No. of heating tubes tested	No. of heating tube damages	No. of leaky heating tubes among them	Sum of heating tube damages (cumula.)	Radiation burden of amount of leakage	Cause of defect, if given	Bibl.
Palisades Inconel 600	PWR 723	1971	2 SG (8519)	1972:80s 1974:100s	1973:1700 1974:1260 1975:285		1973:1700 1974:2660 1975:2945	Aug. 73: Max. leakage limit exceeded 68 l/hr	1973:Thinning of tube walls due to local thermohydraulic conditions in 11 tube rows of the partition in the area of the tube bends, on oscillation damper belts and cross-braces.	[1] [2] [3]
Point Beach 1 Inconel 600	PWR 524	1970	2 SG (3260)	1974:34s	1973:0 1974:2 1975:167	1975:1	1973:193 1974:195 1975:362		1974:Material erosion and intercrystalline crack formation. 1975:Contraction depths on heating tubes of up to 0.1 mm. 1974:Defects in the center of the hot leg bundle in the area of the mud deposits and the periphery. 1975:Uniform erosion due, among other things, to phosphates.	[1] [2] [3] [1] [2] [2]
Point Beach 2 Inconel 600	PWR 524	1972	2 SG (3260)	1974:32s	1973:0 1974:7 1975:3 1976:10	1975:1 1978:1	1973:0 1974:7 1975:10 1976:20	550 l/days of primary water	1975:Defects in the area of the mud deposits over the bottom raft. 1978:Heating tube damage	[3] [6]
S.S. Robinson II Inconel 600	PWR 739	1971	3 SG (3260)	1972:100s	1972:32 1973:3 1974:36 1975:14 Sept.76: 2		1973:35 1974:71 1975:85	1973:15 R/hr within the SG, total: 54 man-rem. 1978:23.5 l/min.	1972:Defects in area of tube sheet. 1973:Intercrystalline corrosion 1 m above tube sheet in hot leg and 1 instance of damage due to erosion in tube bend area. 1975:Uniform erosion in area of mud deposits and in area of bends.	[1] [2] [3] [8]

Figure [sic, should be "Table"] 2.4.1.1. Overview of SG Heating Tube Leaks Throughout the World. [Continuation]

Reactor plant	Type (MW)	Year of start up	No. of SGs heating tubes per SG)	No. of heating tubes tested	No. of heating tube damages	No. of leaky heating tubes among them	Sum of heating tube damages (cumula.)	Radiation burden of amount of leakage	Cause of defect, if given	Bibl.
S.S. Robinson II Inconel 600 [cont'd]									1978:2 heating tubes leaking 1979:Indications of crack formation.	[9]
San Onofre I Inconel 600	PWR 450	1967	3 SG (3794)	1973:50s 1975:1s	1972:23 1973:30 1974:5 1975:19	1975:2	1973:54 1974:59 1975:78	1973:9-10 R/hr on catwalk, total: 50 man-rem.	1972:Defects in area of oscillation damping rods and in first support area. 1973:Constrictions on first cross-brace of tube bundle in hot leg, abrasion in tube bend area. 1974:Damage in bundle area below first support plate. 1975:Fretting corrosion and constrictions in area of oscillation damper rods and support plates. June 1979:Cracks in two of three SGs in the welds of feedwater pipe nozzle.	[1] [2] [3] [9]
Shipping-port Inconel 600	90		4 SG		1973:0		1973:141 1974:141		1974:Leaks (1 SG unusable). June 1979: Signs of crack formation.	[2] [9]
Surry I Inconel 600	PWR 824	1972	1 SG (3388)	1974:77s	1974:143 1975:301	1975:2	1974:143 1975:444	May 74: leakage. Fall of 74 leakage.	1974:Defects due to erosion in hot leg. 1975:Uniform erosion in area of mud deposits.	[2] [3]

Figure [sic, should be "Table"] 2.4.1.1. Overview of SG Heating Tube Leaks Throughout the World. [Continuation]

Reactor plant	Type (MW)	Year of start up	No. of SGs heating tubes per SG)	No. of heating tubes tested	No. of heating tube damages	No. of leaky heating tubes among them	Sum of heating tube damages (cumula.)	Radiation burden of amount of leakage	Cause of defect, if given	Bibl.
Surry II Inconel 600	PWR 824	1973	3 SG (3388)	1974:71s	1974:58 1975:132		1974:58 1975:190		1974:Erosion in hot leg. 1975:Uniform erosion in area of mud deposits. 1979:Cracks and signs of crack formation.	[2] [3] [9]
Takahama 1 Inconel 600	PWR 826	1974	3 SG (3388)		1974:0 1975:98		975:98		1974:Uniform erosion in area of support plates.	[2] [3]
Turkey Point III Inconel 600	PWR 728	1972	3 SG (3260)		1973:0 1974:34 1975:0	1974:2	1974:34 1975:34		1974:Corrosion in center of tube bundle near tube sheet and at first support.	[2] [3]
Turkey Point IV Inconel 600	PWR 728	1973	3 SG (3260)		1973:0 1974:159 1975:5		1973:0 1974:159 1975:164		1974:Corrosion in center of tube sheet in vicinity of tube sheet and at first support. 1975:Defects at periphery of tube bundle.	[2] [3]
Prairie Island 1	PWR 540	1973				1979:1		Evacuation of power house due to boundary value being exceeded.	1979:Rupture of a SG heating tube due to entrained piece of metal - reactor shut down.	[11]

Figure [sic, should be "Table"] 2.4.1.1. Overview of SG Heating Tube Leaks Throughout the World. [Continuation]

Among other things, the table indicates the following:

1. Contradictory data in columns 6 and 8 are due to different statements in the literature.

2. The causes of the SG damage are generally not proven, or are incompletely proven.

3. Leaks occur both with Inconel 600 tubing and with Incoloy 800 tubing.

4. The heating tube proportion relatively most frequently checked is less than 10% of the total number of heating tubes.

5. The leakage rate is generally not given. The data do not allow any conclusions with regard to the types and sizes of the leaks or regarding the amounts which were leaked.

In summary, it can be established that not enough heating tubes are examined for defects. It is to be assumed that if all heating tubes were regularly checked, the number of instances of damage detected and thus the possibility of preventing a heating tube leak would be greater.

2.4.2. Analysis of the Causes of SG Heating Tube Leaks

A precise analysis of the SG heating tube damages shows that the leaks which occurred are less dependent on the material of which the heating tubes are made, but rather are dependent on the water chemistry [18]. In this connection, the occurrence of evaporator overhead condenser leaks (compare [27]) through which the river or sea water gets into the secondary circuit plays a decisive role. In addition to the water chemistry, the consequences of mechanical actions of foreign objects (for instance entrained metal parts), transients and human failure are also important causes of SG heating tube damage.

By way of an overview, the causes of SG heating tube leaks can be divided into the following four groups:

-- 1. SG heating tube leaks caused by corrosion: Here the water chemistry plays a decisice role. Essentially this involves stress crack corrosion, surface eroding and thus wall-reducing corrosion (abrasive corrosion, wastage), pitting corrosion and corrosion in the area of the tube retainers (tube constriction, denting). Since there are many summary works on the problems of corrosion, for instance [17], [19], [28], no further statements will be made here.

-- 2. SG heating tube leaks due to mechanical effects of foreign objects (for instance, detached metal parts): In the history of nuclear power operation, there are many cases of foreign objects of the most widely varying type being found in reactor plants, compare [29], [21], [32]. The new standard SGs of the KWU (Figure 2.4.2) are particularly endangered since 90% of the feedwater is fed at the level of the SG heating tubes. Foreign objects can strike the SG heat tubes directly (and horizontally) and lead to leaks at that point which fall into the category of small LOCAs.

On October 2, 1979 in the U.S.A. the "Prairie Island 1" NPP experienced an incident of this type. A metal coil 20 cm long and 2.5 cm in diameter

(origin unknown) damaged three SG heating tubes; a crack 4 cm long developed in one heating tube [29]. So much activity was conveyed into the secondary loop that the power house had to be cleared becaused the boundary value was exceeded.

Detached metal parts are also known in Germany NPPs. In Biblis A, for instance, it was noted during an inspection that a guide blade of the high-pressure turbine had been broken off during operation. In the description of the inspection and the associated repairs on the high-pressure section of the turbine, it is stated ([21], p. 185 [foreign text]):

> "After opening the high-pressure section in the manufacturer's shop, we were obliged to note that in the first row of guide blades one blade was broken out, but it could no longer be found. In addition, other rivet pins in the first two guide blade rows of the double-flow high-pressure turbine were cracked or torn away..."

In this connection, let us also cite the unexplainable flapping noises in the Biblis A NPP prior to the shutdown on 2/2/79. The studies made during the inspection were unable to clarify the cause of the flapping noises in the plant's primary system [22].

-- 3. SG heating tube leaks due to human error: Tools left lying around after repairs in NPPs are a well-known phenomenon. The consequence of entrained tool parts can be a SG heating tube rupture as described in item 2.

As another example, let us cite an extremely large rise in the chloride concentration in the feedwater due to human error. This can cause, among other things, stress crack corrosion with SG heating tube ruptures. A corresponding case has already occurred in the history of Germany reactors: in cleaning the gravel bed filters in the AVR-Juelich [AVR = Test Reactor Working Association, Incorporated] high-temperature reactor, there was an accidental injection of HCl into the secondary water. The extremely high chloride concentrations in the feedwater caused heavy damage to the SG heating tubes so that major leaks occurred (May 1978). Subsequently the reactor had to be shut down for more than one year for repair work.

-- 4. SG heating tube rupture due to transients: In this connection, it must be stated that predamaged SG heating tubes cannot always be detected by the eddy current testing used in inspections. Because of the slight mass erosion involved, instances of damage which are caused by, for instance, pitting can only be detected when the corresponding point on the heating tube wall has already been reduced by 70-80%. This fact means that even when 100% of the SG heating tubes are checked by eddy-current inspections, many undetected weak points may exist.

In addition, the walls of the SG heating tubes are weakened by material erosion as operating time increases. In 1980 in Biblis A and B approximately 99% of the SG heating tubes had 10-20% wall erosion, 1% had wall erosion of 20-40%, and three tubes were pulled because wall erosion of more than 50% was detected.

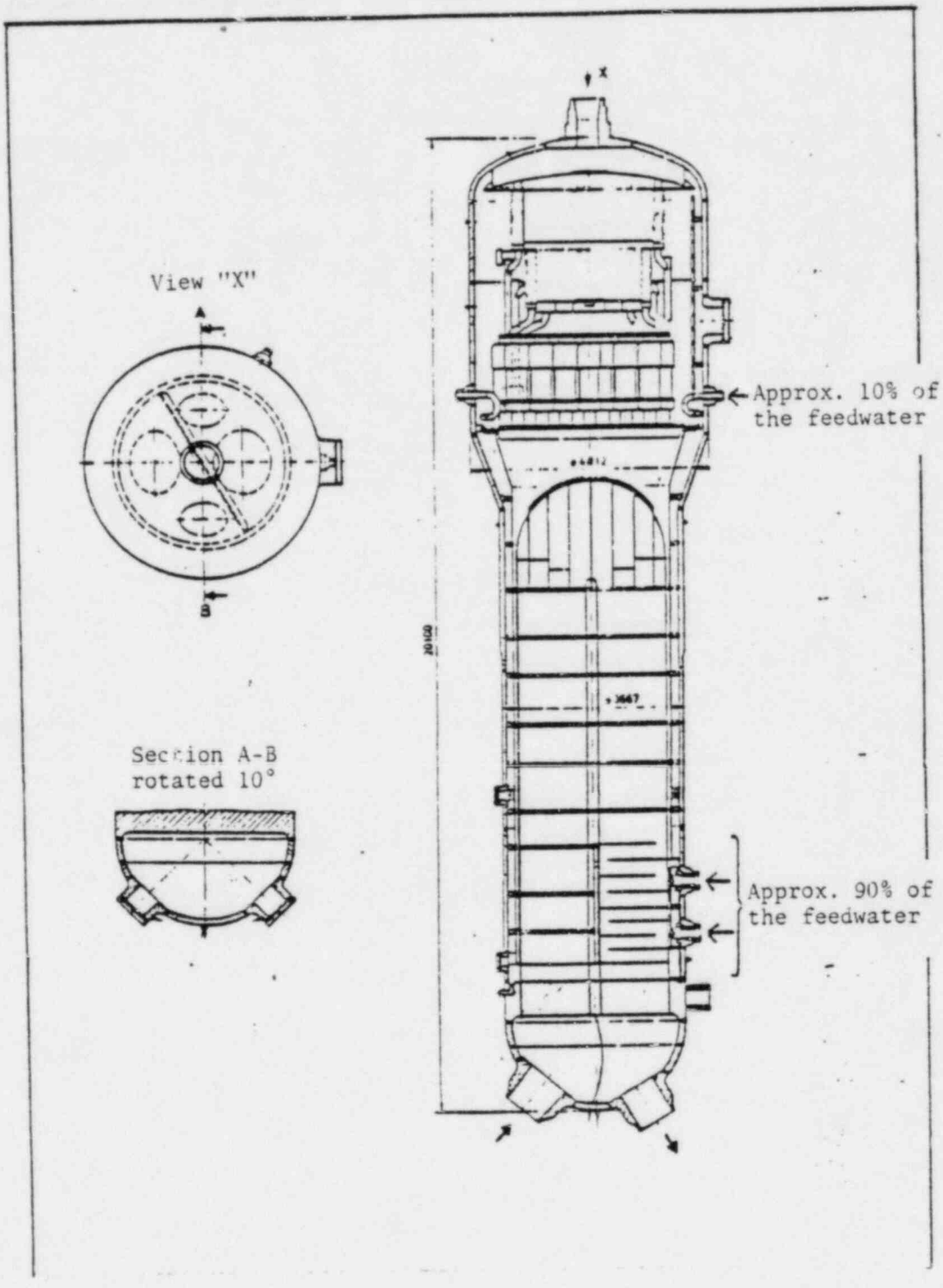


Figure 2.4.2. Standard Steam Generator 54 GS.

According to: Riess, R. and H. J. Schroeder: Chemical Operating Experience in Several European Pressurized Water Reactor Plants. KWU (publisher), 1979.

Reduced wall thickness, hidden pitting corrosion and other corrosion processes increase the probability of the occurrence of SG heating tube leaks during transients. This means, for instance, both small pressure transients which frequently occur during operation and also large pressure transients such as occur in the event of a feedwater pipe rupture. Particularly vulnerable are the new standard SGs of the KWU which are shown in Figure 2.4.2 since the transients can be transferred directly to the heating tubes by means of the medium water. According to our data, Westinghouse discontinued development of similarly designed SGs because excessive SG heating tube leaks due to transient effects were feared.

During major transients caused, for instance, by a feedwater pipe rupture, damage to more than one SG is to be expected. Since the four SGs represent a communizing system, with transit times in the millisecond range the shockwaves underrun every isolation valve, in particular the check valves during the rest of the course of the accident. The radioactive emissions which are thus caused can be a multiple of the emissions which were previously calculated for the "design accident" feedwater pipe rupture. The quotation marks around the word design accident indicate that the effects of the above-designed sequence may extend far beyond a design accident.

In summary it can be stated that regardless of the material used and the design of the SGs, it is impossible to rule out SG heating tube ruptures during the operation of a PWR. This statement applies not only to large transients (such as due to a main steam pipe rupture or feedwater pipe rupture). Cases are known in which spontaneous SG heating tube failure occurred after several years of operation without being caused by large transients (compare Oconee 1976 [3] and Ginna 1982 [33]). This knowledge requires that the accidents taken into account to date within the framework of a guideline to § 28, paragraph 3, of the Radiologic Protection Ordinance be expanded.

2.4.3. The N-16 Signal in the Secondary Loop, a Weakpoint in the Reactor Protection System and Thus the Safety System of a PWR

2.4.3.1. The Scram Limit Value of the N-16 Signal is Not Linked to Reactor Power

It is assumed today that approximately 72% of the core meltdown accidents will be caused by small LOCAs ([23], Figure 3). At least since the accident in the Harrisburg NPP, the significance of such event consequences has been undisputed in technical circles. To date little attention has been paid to small LOCAs which are caused by SG heating tubes ruptures, as well as their consequences.

SG leaks are inadequately recorded by the N-16 signal since this signal is not tied to the reactor power; on the other hand, the generation of N-16 in the reactor is directly dependent on the neutron flux density and thus on reactor power. The nitrogen isotope N-16 is formed from the oxygen of the primary water O-16.

$$O\text{-}16\ (n, p)\ N\text{-}16.$$

In the first approximation, it can be assumed that the formation rate of N-16 at full load will be twice as high as at 50% load. Since the N-16 signal detects SG leaks in the secondary loop only by means of N-16 decays and has a fixed scram limit, according to E. Schruefer ([24], Table 2/23) this means

At 100% load, scram at 10 t/hr SG leakage
At 50% load, scram at ~20 t/hr SG leakage
At 30% load, scram at ~33 t/hr SG leakage

These leakage quantities already lie within the range of a very small to small LOCA (compare [23], [29]) although the response value of the N-16 scram has not quite yet been reached, i.e., the reactor remains in operation. The implications which this has for secondary loop emissions will be discussed elsewhere.

2.4.3.2. Observance of the Maximum Permissible I-131 Fresh [as in text, word probably omitted, should be "Main Steam"] Cannot be Guaranteed by the N-16 Signal

In order to limit the effects of SG leaks, there are officially stipulated maximum I-131 main-steam concentrations of:

10^{-7} Ci/t during pasture [grazing] times

10^{-6} Ci/t at other than grazing times

(compare D. Holm [25], Table 1: Mean Permissible I-131 Activity Concentration in the Secondary Loop 5×10^{-7} Ci/t).

Observance of this maximum I-131 main steam concentration is not guaranteed by the reactor protection system (N-16 signal). The concentration of N-16 in the main steam depends primarily on the neutron flux density (reactor power) and the size of the SG leak. The I-131 concentration depends above all on the burn-up, the number of defective fuel element claddings, the size of the SG leak and the decofactor in the SG. It is therefore conceivable for the N-16 concentration in the main steam to be large relative to the N-16 scram limit with the I-131 concentration being small, and vice versa.

This has consequences for the maximum operating leaks in the SG which can be tolerated. Unlike the employees of the KWU [26], we cannot assume that the N-16 scram will limit the I-131 concentration in the main steam to the values stipulated by the Federal Minister of the Interior.

For the same reason, attempts to determine maximum operating leaks by calculating back from the maximum I-131 main steam concentration (for instance [25], page 5; [30] A/4ff) are to be rejected. In addition, regarding these calculations it should be noted that, depending on the parameter values used, completely different leakage values are calculated. Thus, for instance, it is important to know with what throughput and with what decontamination factors the blowdown demineralization system of the SGs operates. Another parameter is the decofactor in the SG; according to the present state of the art, it is not permissible to assume that transport occurs with residual moisture only. Check calculations show that a SG leak calculated from the maximum main steam concentration of I-131 can fluctuate by a factor of more than 1000 (from several kg/hr to several to t/hr). From this standpoint as well, these back calculations are unsuitable for determining maximum operating leaks.

The maximum operating leakage must be postulated to be a value which just barely does not initiate the N-16 scram. In our case, this is thus approximately 10 t/hr at 100% load. This is also required by KTA rule 3501 (10/80). In 3.3. "Initial State of the Plant", it says:

> "For any event sequence, the operating state of the plant which is least favorable with regard to its effects must be assumed."

It is another question how long such a leak can continue with the reactor in operation.

Since the leakage is recorded in the measurement devices of the SG blowdown system and the evaporator overhead condenser exhaust system as well as by the N-16 detectors, it is to be assumed that relatively quickly the measurements of the I-131 activity in the main steam which are planned for this case will be undertaken. If the measure value of the I-131 concentration is above the officially established value, the cognizant authority must be notified, and this authority will then react by issuing instructions to the operator in accordance with the situation. We estimate that this process will take somewhere between several days and eight weeks.

2.4.4. Determination of SG Leaks for the Parameter Study

The statements on the amount of leakage which leads to reactor scram by means of the N-16 signal are widely divergent. E. Schruefer [24] indicates a value of 10 t/hr, the safety report on the Grafenrheinfeld NPP cites a value of 3 t/hr, and one operator gave us a value of 0.4 t/hr, all at 100% reactor power. In addition to the scram limit of the N-16 device, these quantities of leakage depend primarily on the reactor power and on the type and location of the leak (cold leg, hot leg, tube bend, above the SG plate).

If for the sake of conservativeness we adopt the value of E. Schruefer as the boundary value for the maximum operating leak at which the N-16 scram barely avoids activation, then the following dependency on reactor power results:

Reactor power for a 1300 MW/e reactor	Maximum amount leaked without a N-16 scram
100%	10 t/hr
50%	20 t/hr
30%	33 t/hr

In the parameter study the SG leak was varied from 2×10^{-2} (t/hr) to 10 (t/hr).

Since corrosion is not a linear process and eddy current testing does not completely record corrosion damage to the SG heating tubes, leaks of more than 10 t/hr cannot be ruled out, particularly in the case of NPPs with longer operating times. The possible causes were presented in 2.4.2. From this it is reasonable to carry out emission calculations with larger leakage quantities. In this process a reactor scram occurs (the N-16 signal). This accident leads to particular problems since usually an intact secondary loop is assumed for after-heat removal in the case of a small LOCA; in this case, however, it is specifically this which is not given. A core

meltdown accident with secondary-side emissions may be the consequence.

In any case a large SG heating tube leak may lead to activation of the safety or relief valves and thus to massive releases of activity. Even if in the case of a N-16 reactor scram a delayed turbine trip is provided for, activation of the safety or relief valves cannot be ruled out.

2.4.5. Bibliography for 2.4

1. IRS Brief Information 1976/A/4 and 1976/A/5.

2. IRS Brief Information 1977/A/6.

3. IRS Brief Information 1978/A/19 and 1978/A/21.

4. Schenk, H., VGB KRAFTWERKSTECHNIK, Vol. 4, 1977.

5. GRS Brief Information 1977/A/7.

6. GRS Brief Information 1978/A/62.

7. GRS Brief Information 1979/A/27.

8. GRS Brief Information 1979/A/15.

9. GRS Brief Information 1979/A/53.

10. BMI: Overview of Special Events in NPPs of the FRG in 1979. Bonn, 1980.

11. GRS Brief Information 1979/A/68.

12. KKS [Stade NPP] - Leak at a SG Heating Tube. Report to the RSK, September 1981.

13. Personal communication from employees of the Kraftanlagen Company, Heidelberg.

14. Letter from the Administrative Court of Baden-Wuerttemberg to Attorney Rainer Beeretz dated 11/13/80.

15. Letters from Attorneys Dr. Elizabeth Ziegert Ulrich Ziegert and Dr. Hans Schroedter to Attorney de Witt dated 10/21/80.

16. Personal communication of the operator on 6/12/81 in response to previous written request dated 5/18/81.

17. Stevens Guille, P. D., "Steam Generator Tube Failures: A World Survey of Water-Cooled Nuclear Power Reactors to the End of 1971", ATOMIC ENERGY OF CANADA LIMITED.

18. Schenk, H., Dr., Obrigheim NPP, "Overview of Heating Damage in SGs of NPPs", 4th MPA [Materials Testing Institute] Seminar, October 4-5, 1978, "Rupture Behavior and Rupture Phenomena - Primary System and Containment".

19. Guelich, J. F., "Importance of Thermohydraulic Conditions for the Corrosion Safety of PWR Steam Generators", ATOMWIRTSCHAFT, pp. 82-87, February 1975.

20. NUCLEAR NEWS, pp. 35-38, November 1979.

21. Spalthoff, F. J., "First Inspections of Unit A of the Biblis NPP", AED Conference, pp. 183-200, 1976.

22. GRS: "Expert's Position on the Planned Monitoring Measures Upon Restart and in Subsequent Operation of the Biblis A NPP After the Shutdown on 2/2/79", Cologne, 1979.

23. Farber, G. and F. Wolpert, GRS: "Analysis and Control of Small Breaks in German PWRs and Accompanying Research Activities", IAEA, October 20-24, 1980.

24. Schruefer, E., "Strahlung und Strahlungsmesstechnik in Kernkraftwerken" [Radiation and Radiation Measurement Technology in NPPs], Berlin, 1974.

25. Holm, D., "Theoretical Determination of Noble Gas Nuclide Spectra in the Exhaust Air of NPPs and PWRs", GRS Report 8, 1978.

26. Stuenkel, H., G. Hecht and E. Doerr, "Retention of Radionuclides", Technical Conference on Radioecology, German Atomic Forum, Bonn, October 2-3, 1979.

27. Schroeder, H. J., "VGB Feedwater Conference", 3/79, p. 195 ff.

28. Baschek, H. and E. Kocourek, "Operational Experience with Heat Exchangers in Nuclear Power Stations with Light Water Reactors", COMBUSTION, pp. 14-21, September 1975.

29. Tong, L. S., "International Conference on Current Nuclear Power Plant Safety Issues", IAEA, Stockholm, 10/20-24/80, USNRC LOCA Research Programs.

30. Hannover TSA, "Response to and on Items CV-CIX of the Grounds for Complaint in the Administrative Law Cases Involving the Grohnde NPP", Hannover, February 1978.

31. Personal communication from employees of the Kraftanlagen Company.

32. "Action Reports from the Federal Government" published in: "Unfaelle in Deutschen Kernkraftwerken" [Accidents in German NPPs], Federal Association on Citizens' Initiatives on Environmental Protection, registered association, 1979.

33. NEWSWEEK, "Again a Nuclear Mishap", p. 43 ff., February 8, 1982.

2.5. Decontamination Factors

- Introduction

Decontamination factors (DF) occur at many points in the secondary circuit. They are significant for nuclide transport. The decontamination factors of "typical salts" (for instance, sodium salts, magnesium salts) differ greatly from the DFs of typical gases (for instances, noble gases). The nuclide transport at the individual points (for instance, the SG) is determined by the residual moisture and by direct transfer with the steam. Measurements by Hood [1], Schroeder et al. [2] and Jonas [3] show that direct transfer of radionuclides with steam is observed not only in the case of noble gases, but also in that of fission products (for instance, Cs 137, I 131) and activation products (for instance, Co 58).

These experimental observations contradict the statements of Neeb et al. [4] as well as corresponding assumptions in the expert opinion from the Hannover TSA [5]. In both publications it is assumed that, for instance, I 131 is transported in the SG exclusively with the residual moisture. In its expert opinion [5], the Hannover TSA even assumes that in the SG all fission and corrosion products, with the exception of the noble gases, are transported exclusively with the residual moisture. This false assumption leads to an underestimate of the secondary emissions which are calculated there.

The works of Styrkovich and Chaybullin [6] and that of Leibovitz [7] show that during the phase transition of all salts a certain portion is transported directly with the steam. Here this "steam carryover" is mainly conceived to be the transport of neutral molecules from the liquid phase into the vapor phase. Our conventional concepts of assuming intense electrolysis in water as completely dissociated must be reviewed. Under reactor conditions at 285°C in the liquid phase, even Na Cl itself is a weak electrolyte and is not completely dissociated. In general, it can be stated that both the degree of dissociation and the "steam carryover" are significantly dependent on all other escort substances, in addition to temperature and pressure. Even today we know very little about the precise processes which take place under reactor conditions, for instance in the SG. One thing is, however, clear:

Through the interface in the SG, there is nuclide transport both via the residual moisture and directly via the dry steam.

At this point let us mention a quote from Jonas ([8], page 456 [foreign text]) which summarizes many of the existing gaps in our knowledge about the transport of elements in the SG and other plant parts as follows:

> "It is no contradiction that there are precipitates on the turbines in PWRs although the purity of the steam seems to be satisfactory. Possible causes thereof are:
>
> a). The composition of the steam is considered only for steady-state operation -- during small transients the conditions are unknown.
>
> b). The presence of salts with very low solubility in dry steam.
>
> c). The interaction of components which are dissolved in the dry steam with metal surfaces."
>
> (Translation by the IFEU, 1981)

The following section discusses the experimental data which underlie the decontamination factors in the SG and the bandwidths used in the parameter study. The decontamination factors at the various points are calculated by the computer program from the residual moistures and the factor α which is decisive for direct steam transport (α is the ratio of the specific activity in the liquid phase to the specific activity in the steam phase). The formulas used are presented in Section 2.2 (compare in particular equation (18)).

In addition, conservative and mean α values for different nuclide groups are derived for the SG on the basis of the experimental data.

2.5.1. Decontamination Factors in the SG

The design residual moisture in the main steam behind the steam purifiers of the SG is indicated as being 0.25% in the case of the KWU steam generators of the Biblis type. However, this value is applicable only in the case of constant load operation. The work of Dibelius et al. [9] demonstrates how difficult it is to verify this specification value experimentally. In the acceptance tests for Biblis A, mean values of the residual moisture of up to 0.32% were determined in measurement with the scattered-light probe. By using other physical measurement methods, an attempt is being made to make it clear that the specification value of 0.25% can be observed.

In the case of transients and in the case of SG leaks, the residual moisture can be considerably higher. In the parameter study an interval of 0.1% to 31% was thus selected for the residual moisture.

To some extent the values for nuclide transport directly with the steam are given in the literature, and to some extent they can be derived from published measurement data. In this case the "vapor transfer factors" (VTF) obtained according to Hood [1] are only slightly different from the α values used in the computer program (compare equation 8, Section 2.2).

Measured DFs for Na, I 131 and Cs 137 are published in the work of Hood ([1], compare page 20 ff [foreign text]). The following mean $\bar{\alpha}$ values and minimum α_{min} values can be obtained.

	I 131	Cs 137
$\bar{\alpha}$	860	730
α_{min}	467	260

We consider these results transferable to the SGs used in the FRG.

From the work of Westinghouse [14] the following minimum and mean α values for I 131 can be obtained for the SG.

	I 131
$\bar{\alpha}$	1050
α_{min}	270

In Figure 3 (see Section 2.5.1.1) Jonas [3] presents the distribution of different chemical substances between the steam and the water as a function of pressure. An analysis of the graphs with respect to α values at approximately 54 bar (pressure in the SG) yields α values > 10^6 for Na salts. This result is quite consistent with the works of Class [10], [11], [12], [13], from which VTF values or α values of > 10^6 can be derived for Na salts.

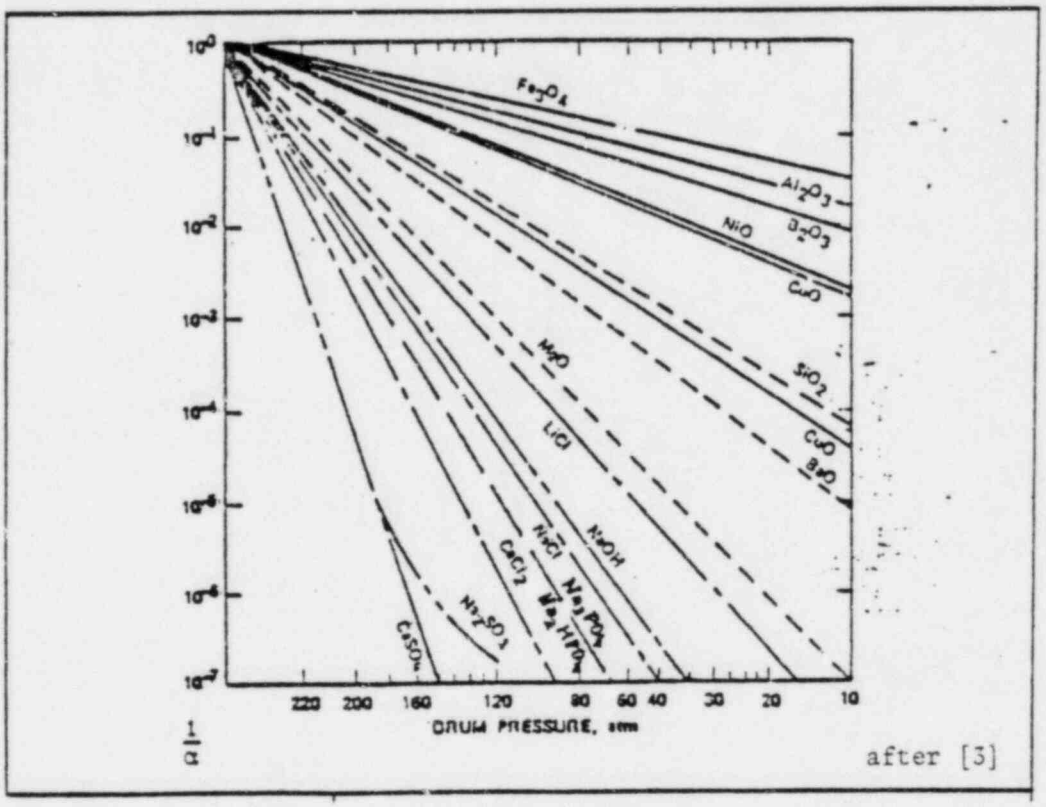


Figure 2.5.1.1. Distribution of Chemical Substances Between the Steam and Water Phases as a Function of Pressure.

The following Table 2.5.1.2 lists the α values for compounds of iron, aluminum, boron, nickel, copper, silicon, varium, magnesium, lithium, sodium and calcium according to Jonas [3]. Reading errors result in an error range of $\Delta\alpha \pm 12\%$.

TABLE 2.5.1.2. α VALUES AT 54 BAR ACCORDTNG TO JONAS [3].
DUE TO READING ERRORS FROM THE GRAPHS Δα ± 12%.

Chemical substance	α Δα ± 12 %
Fe_3O_4	10
Al_2O_3	16
B_2O_3	25
Ni O	64
Cu O	75
$Si\ O_2$	69(+1)
Ba O	27(+2)
Mg O	51(+3)
Li Cl	14(+4)
Na OH	22(+5)
Na Cl	53(+5)
Na_3PO_4 $Na_2H\ PO_4$ $Ca\ Cl_2$ Na_2SO_4 $Ca\ SO_4$	>1(+7)

$$\alpha = \frac{\text{Specific activity in water}}{\text{Specific activity in steam}}$$

$$5(-4) = 5 \times 10^{-4}$$

An analysis of the work of Jonas [3] shows that, for instance, Ba O is transported 2,000 times better with the dry steam than Na Cl, and Fe_3O_4 is transported 530,000 times better with the dry steam than Na Cl.

An analysis of the work of Leibovitz [7] yields the α values listed in Table 2.5.1.3. Due to reading errors, the error range of $\Delta\alpha \pm 31\%$. With the exception of Li Cl, Na Cl and Na OH, the error bars [this word may refer to the "bars" in a bar graph] of the individual α values overlap with the values of Jonas.

TABLE 2.5.1.3. α VALUES AT 54 BAR ACCORDING TO LEIBOVITZ [7]. DUE TO ERRORS IN READING FROM THE GRAPHS, $\Delta\alpha \pm 31\%$.

Chemical substance	α $\Delta\alpha \pm 31$ %
Co O	22
B_2O_3	22
Si O_2	57(+1)
Li Cl	62(+3)
Na OH	48(+4)
Na Cl	25(+5)
Ca Cl	1(+8)
Na_2SO_4, Ca SO_4	3(+12)

$$\alpha = \frac{\text{Specific activity in water}}{\text{Specific activity in steam}}$$

$5(-4) = 5 \times 10^{-4}$

In the works of Jonas [3] and Leibovitz [7] there is a clear tendency for chemical compounds to be better transported with steam the less water-soluble they are under normal conditions. In an extension of the line of reasoning of Liebovitz, even under normal conditions little-soluble substances should be even more difficult to dissolve under reactor conditions. The recognition of good steam transport of little-soluble substances is quite consistent with the concept that the transfer of neutral molecules from the liquid into the steam underlies the transport which takes place via the steam.

From the work of Schroeder et al. [2], the following values for α can be derived for I 131, I 133, Co 58 and Co 60:

	α
I 131	$1{,}25 \times 10^4$
I 133	$2{,}16 \times 10^3$
Co 58	$1{,}25 \times 10^3$
Co 60	$1{,}39 \times 10^3$

$$\alpha = \frac{\text{Specific activity in water}}{\text{Specific activity in steam}} \text{ at 50 bar after [2].}$$

The α value for I 131 differs by a factor of 40 from the values which Hood [1] found, and the α values for Co even deviate by a factor of 63 from those found by Leibovitz [7]. An important reason for the different results could be the fact that Schroeder et al. [2] carried out the measurements in a BWR, while Hood [1] bases his work on a PWR.

In contrast to all previously cited works is a publication from the KWU [15]. In this work, which is flawed from both the theoretical and the experimental standpoints, the KWU attempts to claim that I 131 is transported only with the residual moisture. The work proceeds from the basically false assumption that the chemistry of iodine under normal conditions can be directly transferred to the chemistry of iodine under reactor conditions. In accordance with this assumption, samples taken from the reactor under normal conditions are examined with regard to their chemistry, and then the false conclusion is drawn that things are exactly the same under reactor conditions. By the same token, theoretical considerations which apply to the chemistry [of the element] under normal conditions are directly transferred to reactor conditions without critical examination. We quote:

> "From the transfer of the known chemical properties of iodine to the conditions prevailing in a PWR with a natural-circulation SG, it is thus found that the transfer of fission iodine into the steam should occur only by mechanical means with the moisture of the steam."

The subjunctive "should" becomes an indicative at the end of the work. Thus, the conclusion states:

> "The transfer of fission iodine into the main steam can thus occur only by mechanical means through the steam moisture when the water chemistry meets the specifications. The iodine activity which gets into the main steam when there is an SG operating leak is thus unambiguously limited by the transfer rate which is proportional to the steam moisture and by the steam moisture which is specificed to be a maximum of 0.25%."

The experimental residual moisture determinations in the SG by which the above conclusion is supposed to be buttressed experimentally are based on another false assumption. It is assumed that Cs 137 is transported only with the residual moisture. This assumption is refuted by, for instance, the work of Hood [1].

It is regrettable that this inaccurate publication of the KWU and the false conclusions drawn from it are still being incorporated even today (1981) into the calculations of the GRS [16] and the KWU]17] for secondary loop emissions. In addition, observance of the maximum main steam concentrations of 10^{-6} Ci/t of I 131, as prescribed by the BMI, is being monitored by the operators under normal conditions on the basis of the false assumption [this element] is transported by residual moisture alone.

Since the direct transport of the individual nuclides by steam fluctuates a great deal, the α values were varied from 10^{-4} to 10^{5} in the sensitivity analysis [Section 4). A α value of 10^{-4} corresponds to the transfer of a noble gas, $\alpha = 1$ corresponds to the transfer of tritium in the chemical form of HTO, and the α values of $10\text{-}10^{5}$ correspond to the transfers of saline substances.

In Section 5, nuclide-specific α values are used. For this purpose the nuclides were subdivided into groups, which largely coincide with the group division made by the Baden TSA [18]. Since very few experimental data are available and since at the same time there are many unsolved problems (for instance, that of the chemical form of the nuclides under reactor conditions), α values and mean α values were estimated for the individual nuclide groups on the basis of the above-cited experimental data (Table 2.5.1.1). No measured values were available for the "actinides" group [18] (the quotation marks indicate that, in addition to actinides, this group also contains other elements). Since, however, it is to be expected that these elements are partially present in unsoluble form under reactor conditions, α values which are similar to those for the group "other solids" are to be expected. We have therefore consolidated the two groups. Elements which fall into the group "activation products" also sometimes have isotopes from fission products, and therefore they may be counted more than once. The activation product Cl was included in the halogens group, and the activation products Th and Pa as well as the transuraniums were assigned to the group "other solids". On the basis of the available data material, a more precise breakdown was not reasonable.

For compounds of the elements sodium and calcium, α values in excess of 10^{6} were measured. For isotopes of these elements, it is therefore reasonable to assume transport by residual moisture alone although traces of these elements are still transported via the steam.

2.5.2. Decontamination Factors (DF) in the Filters of the SG Blowdown Demineralization System

The nuclide-specific DFs in the filters of the blowdown demineralization system fluctuate a great deal. In the sensitivity analysis, these DFs are thus varied between 1 (noble gases, tritium) and 10^{4}. In the works of the GRS, a mean DF of 100 is calculated [19].

2.5.3. Other Decontamination Factors

The corresponding DFs at the various other points are determined by the computer program using the data summarized in 2.6 according to the formulas derived in 2.2.

Nuclide group	$\alpha_{conservative}$	$\alpha_{mean\ value}$
I Noble gases	10^{-4}	10^{-4}
(Isotopes of Kr, Xe)		
Of particular importance are the decay products of short-lived noble gases, of which we should mention here Sr 89, Sr 90, Sr 91, Ce 93, Ce 95, Cs 135, Cs 137, Cs 141, Nd 143 and La 140, as well as other of their daughter products.		
II Halogens	260	800
(Isotopes of Cl, Br, I)		
III Solids which are volatile under 1,200°	250	700
(Isotopes of Se, Rb, Te, Cs)		
IV Other solids	500	3,000
(Including actinides)		
(Isotopes of Zn, Ga, Ge, As, Sr, Y, Zr, Nb, Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, Sb, Ba, La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, U, Np, Pu, Am, Cm, Th, Pa)		
V Activation products	10	400
(Isotopes of Cr, Mn, Fe, Co, Zr, Ni, Mo, W, Sb, Be, Al, S, P, S, Sc, V, Nb, Tc, Sn, Te, Ta, Na, Si, Ca, Cu, Zn, Sr, Th)		
VI Tritium	1	1
VII Carbon 14	10^{-4}	10^{-2}

Figure 2.5.1.1. [sic] Conservative and Mean α Values for Individual Nuclide Groups.

In most of the calculations performed, the nuclide-specific α values in the low-pressure section are obtained from the α values of the SG by multiplication by a factor of 100. The main cause of the other α values in the low-pressure section is the different pressure conditions.

In the sensitivity analysis (Section 4) the ratio $\frac{\alpha \text{ low pressure}}{\alpha \text{ high pressure}}$ is also varied between 1 and 1,000. All noble gases which get into the evaporator overhead condenser are evacuated via the condenser exhaust (to the vent stack) and are thus removed from the loop.

2.5.4. Bibliography for 2.5

1. Hood, G. W., "Steam Side Iodine Transport Study at Point Beach Unit No. 1 of Wisconsin Electric Power Company", Wisconsin Electric Power Company, WCAP-8215, 1973.

2. Schroeder, H. J., W. Schermann and G. Class, "Steam Purity Measurements in the Grundremmingen NPP", VGB Feedwater Conference, pp. 11-15, 1967.

3. Jonas, O., "Transfer of Chemicals in Steam Power Systems", ASME Publication 76-JPGC-Pwr-12, The American Society of Mechanical Engineers.

4. Neeb, K. H., H. Richter, E. R. D'Amaddio and R. E. Mayer, "The Behavior of Fission Iodine in the Secondary Loop of a PWR (Summary)", Reactor Conference, F. R. Germany, April 1975.

5. Hannover TSA (publisher), "Response to Items CV to CIX of the Grounds for Complaint in the Administrative Law Cases Involving the Grohnde NPP", The Hannover TSA, KWG SOG 2/1, February 1978.

6. Styrikovich, M. A. and Y. Ch. Chaybullin, "Allgemeine Gesetzmaessigkeiten der Loeslichkeit von Substanzen in Wasserdampf" [General Laws of the Solubility of Substances in Steam], USSR Academy of Sciences Publishing House, translation by: IFEU 1980.

7. Leibovitz, J., "Model of Vaporous Carryover", U.S.A., EPRI-NP-1787, April 1981.

8. Jonas, O., "Transfer of Chemicals in PWR Systems: Secondary Side", Water Chemistry of Nuclear Reactor Systems, BNES, London, pp. 445-457, 1978.

9. Dibelius, G., A. Doerr, A. Ederhof, F. Koziorowski, F. Meier, K. Natusch, E. Ossendorf and W. Schermann, "Experience in the Determination of Steam Moisture in the Acceptance Tests at the Biblis NPP", KRAFTWERKSTECHNIK, Vol. 9, 1977.

10. Class, G., "The Salt Content of the High-Pressure Steam in Industrial Steam Generation", MITTEILUNGEN DER VGB, Vol. 94, pp. 27-36, February 1965.

11. Ibid., "Water Entrainment by Gases or Steam", MITTEILUNGEN DER VGB, Vol. 96, pp. 179-184, June 1965.

12. Ibid., "The Solubility and Phase Distribution of Substances in the Critical State Region of the Solvent", ZEITSCHRIFT FUER PHYS. CHEMIE NEUE FOLGE, Vol. 46, pp. 332-344, 1965.

13. Class, G. and R. Quack, "Practical Calculation of the Distribution of Strong Electrolytes Between Water and Steam", Research work of the Institute for Chemical Engineering and Steam Boilers at the Stuttgart Institute of Technology, Steam Boiler Tube Association, Duesseldorf, October 1963.

14. WCAP 8253, "Source Term Data for Westinghouse Pressurized Water Reactors", U.S.A., May 1974.

15. Naeb, K. H., H. Richter, E. R. Damaddio and R. E. Mayer, "The Behavior of Fission Iodine in the Secondary Loops of PWRs", Reactor Conference at Nuernberg, 1975; KWU and Combustion Engineering Inc., U.S.A. (publishers).

16. Urban, H. and H. Biesold, "Certification of the Radiation Exposure of the Populace by the Release of Radioactive Substances with the Exhaust Air in the Environs of the Site of the Neckarwestheim Community NPP", GRS (publisher), 1978, revised June 1981.

17. KWU (publisher), "Sicherheitsbericht Neckarwestheim II" [Safety Report on Neckarwestheim II], Erlangen, 1980.

18. Baden TSA (publisher), "Activity Release in the Event of Accidents in NPPs with LWRs Via the Exhaust Air Pathway Within the Framework of a Guideline for § 28, paragraph 3, of the Radiologic Protection Ordinance", December 1979.

19. Holm, D., "Theoretical Determination of Noble Gas Nuclide Spectra in the Exhaust Air of NPPs with PWRs", GRS Report, No. 8, 1978.

2.6. Mass Content, Mass Flows and Steam Emissions

- Introduction

The model consideration of the secondary loop provides for determination at 13 points and 21 mass flows between these points (compare 2.1).

For calculating the emissions from the secondary loop using the computer program which is described in Section 3, precise knowledge of the point-related input data (mass content, moisture at that point and dwell time) as well as the flow-related input data (mass flow, moisture of the mass flow and transport time) is required. The Biblis NPP, Unit B, was used as the reference plant or the basis for the input data which were employed.

In the following sections the calculation principles, the data sources and the input data used in the study are presented.

2.6.1. Calculation Principles

Below we will describe the sources, considerations and calculations which underlie the numbers given for the input data cited later.

2.6.1.1. Point-Related Input Data

- Mass Contents of the Points

The basis for determining the mass contents was the data kindly provided to us by the operator on the net or free volumes of the components which were of interest [1].

From the pressure and temperature data which define the steam or water state and which were also provided by the operator, the specific volumes were determined and thus the mass contents were calculated. The mass content of a point indicated in Table 2.6.1 is always the sum of the water and steam masses. The mass content of the total of 12 low-pressure preheaters as well as the total of 6 letdown heat exchangers were added to the hot well (point No. 8). By the same token, the value for the feedwater tank also contains the mass content of the two high-pressure intermediate heat exchangers/condensate coolers, the two high-pressure preheaters and the two high-pressure condensate coolers.

Regarding the data for the blowdown filter, sump and atmosphere, the following remarks should be made:

The complicated absorption conditions of the radionuclides in the blowdown filter were simulated by software means as follows: in a blowdown filter bypass circuit, an amount of water which is defined by the decofactor is fed to the filters past the hot well.* If the decofactor of the filter system is thus set at, for example, 10,000 for a certain nuclide, only 10^{-4} part of the water leaving the flash tank gets into the hot well.

All relevant activity sinks, aside from the natural decay which is due to the half-life are consolidated into the "points" blowdown filter, sump and atmosphere (including condenser evacuation). Their mass contents and their dwell times are not included in the calculations of the computer program. Due to programming considerations, the mass contents of the three above-mentioned points are set at 1 t.

- Moisture of the Mass Content of a Point

The moisture of the mass content of a point is defined as the ratio of the liquid mass content to the total mass content (compare 2.2). The moistures in the SGs and hot well (points 1, 2 and 7) are accordingly calculated from the water contents. Since the mass proportion of the steam in the total mass in the hot well is negligibly small, f = 100% was assumed. By the same token, in the feedwater tank (point 9) the deviation from 100% moisture is also vanishingly small due to the assumed 10% steam volume proportion.

The moistures in the main steam, high-pressure turbine and cyclone separator/ /intermediate superheater are taken from the literature [2], and the moisture of the low-pressure turbine is taken from [3]. The moisture value in the evaporator overhead condenser was estimated from the residual moisture of the low-pressure turbine discharge and a condenser add-on of 5%.

In the flash tank (point 10), the moisture is calculated from the ratio of the masses or the specific volumes of the steam and water contents.

In the blowdown filter and the sump (points 11 and 12) the moisture was set at 100% while in the atmosphere (point 13) it was assumed to be zero.

*In the present case, it is not advisable to define the decofactor as the ratio of the mean concentration of the mass at this point and of the mass flow leaving this point since other significant values such as the filter, charging state and regeneration time/cycle would also have to be taken into account. Instead, the decofactor of the blowdown filters is supposed to indicate the concentration ratio of the mass flows in front and behind the filter.

- Dwell Time of the Activity at One of the Thirteen Points

The dwell times listed in Table 2.6.1 are calculated from the ratio of the mass content M to the mass flow $\dot{m}$ leaving that point according to the formula $t = M/\dot{m}$. If there were branches, the corresponding mass additions or subtractions were consolidated. The dwell time in the feedwater tank (point 9) was calculated on the assumption of a blowdown rate of 4 × 24 t/hr.

The dwell times at the points were taken into account by adding them to the corresponding transport times.

2.6.1.2. Mass Flow-Related Data

- Mass flows $\dot{m}tot$.

The mass flows listed in Table 2.6.1 correspond to the operator's statements [1].

- Moisture of the mass flow $\dot{m}tot$.

The moisture of the mass flow is equal to the ratio of the liquid portion of the mass flow $\dot{m}_{fl}$ to the total mass flow $\dot{m}_{tot}$: $f = \dot{m}_{fl}/\dot{m}_{tot}$.

Let the mass flows between two points x and y be designated by $\dot{m}$ (x, y). The following mass flows are taken from the literature [2]: $\dot{m}_{fl}$(1, 3) (2, 3) (3, 4) (3, 9) (4, 5) (4, 9) (5, 6) and $\dot{m}_{fl}$(6, 7) (6, 8) (6, 9) from [3]. The other moistures were estimated from the thermodynamic state values as well as the operating objectives and, where the liquid phase predominated, f = 1 was assumed.

- Transport times

As is the case with the dwell times at one point, the transport times of the mass flows in the pipes were calculated according to the formula $t = M/\dot{m}$, with $\dot{m}$ = the mass flow through the pipes. Both parallel connection and series connection of the pipes were taken into account in this process.

The very short transit times in the high-pressure section or steam section of typically 10^{-1} sec, in contrast to the transit times of the blowdown pathway of typically 5×10^{3} sec, are significant and decisive for the mass and activity flows.

2.6.1.3. List of the Input Data Used

Table 2.6.1 lists the values of the input data used in the computer program for the point-related data mass content M, moisture of the mass content F and dwell time T and, for the mass flow-related data mass flow $\dot{m}$, moisture of the mass flow f and transport time t. The sources of the values were explained in the previous section.

Input data								
Point	Masses t	Moisture %	Dwell time sec	From	To	Mass flow t/sec	Moisture %	Transport time sec
1	46,900	90.6	95.0	1	3	0.495	0.25	3.40
2	140.700	90.6	95.0	2	3	1.485	0.25	3.40
3	6.986	0.5	3.4	3	4	1.882	0.50	0.14
4	0.033	11.4	17 msec	3	9	0.098	0.50	46.00
5	0.410	11.4	250 msec	4	5	1.625	11.40	0.20
6	0.035	10.0	24 msec	4	9	0.257	11.40	2.30
7	0.029	15.0	26 msec	5	6	1.439	0.25	0.69
8	247.400	100.0	190.0	5	9	0.186	100.00	22.60
9	368.700	100.0	180.0	6	7	1.116	10.00	0.03
10	0.040	66.6	1.5	6	9	0.152	10.00	39.05
11	a)	100.0	a)	6	8	0.171	10.00	183.00
12	a)	100.0	a)	7	8	1.116	15.00	0.00
13	a)	0.0	a)	8	9	1.287	100.00	264.00
				9	1	0.495	100.00	356.00
				9	2	1.485	100.00	356.00
				1	10	b)	100.00	1263.00
				2	10	c)	100.00	1263.00
				10	9	c)	66.66	1.50
				10	11	c)	100.00	2303.00
				10	8	c)	100.00	2303.00
				11	8	c)	100.00	0.00
				3	13	d)	0.50	f)
				9	13	e)	50.00	f)

1 SG leaky
2 SG intact
3 Main steam
4 High-pressure turbine
5 Water separator intermediate superheater
6 Low pressure turbine
7 Evaporator overhead condenser
8 Hot well
9 Feedwater tank
10 Flash tank
11 Blowdown filter
12 Sump
13 Atmosphere

a) The water content and dwell time are not included in the computer program.

b) Blowdown rate is variable, 0.0-24 t/hr per SG.

c) The values are designated from the mass flow 1.10.

d) Steam leakage from the high-pressure section is variable 1-10 t/hr.

e) Steam leakage from the feedwater degasification system is variable 0-6 t/hr.

f) The values are calculated from d) and e).

Figure 2.6.1. Input Data Used in the Computer Program for the Point-Related and Mass Flow-Related Data.

In Section 4 (the parameter study) the residual moisture was also varied between 0.01% and 31.2%.

2.6.2. Mass Flows When There is a Load Change

For calculations of emissions during load change processes, mass flows as measured during the acceptance measurements of Biblis, Unit B, are used (compare Figure 2.6.2).

The mass contents, mass flows, dwell times and transport times which correspond to 100% power are shown in Table 2.6.1.

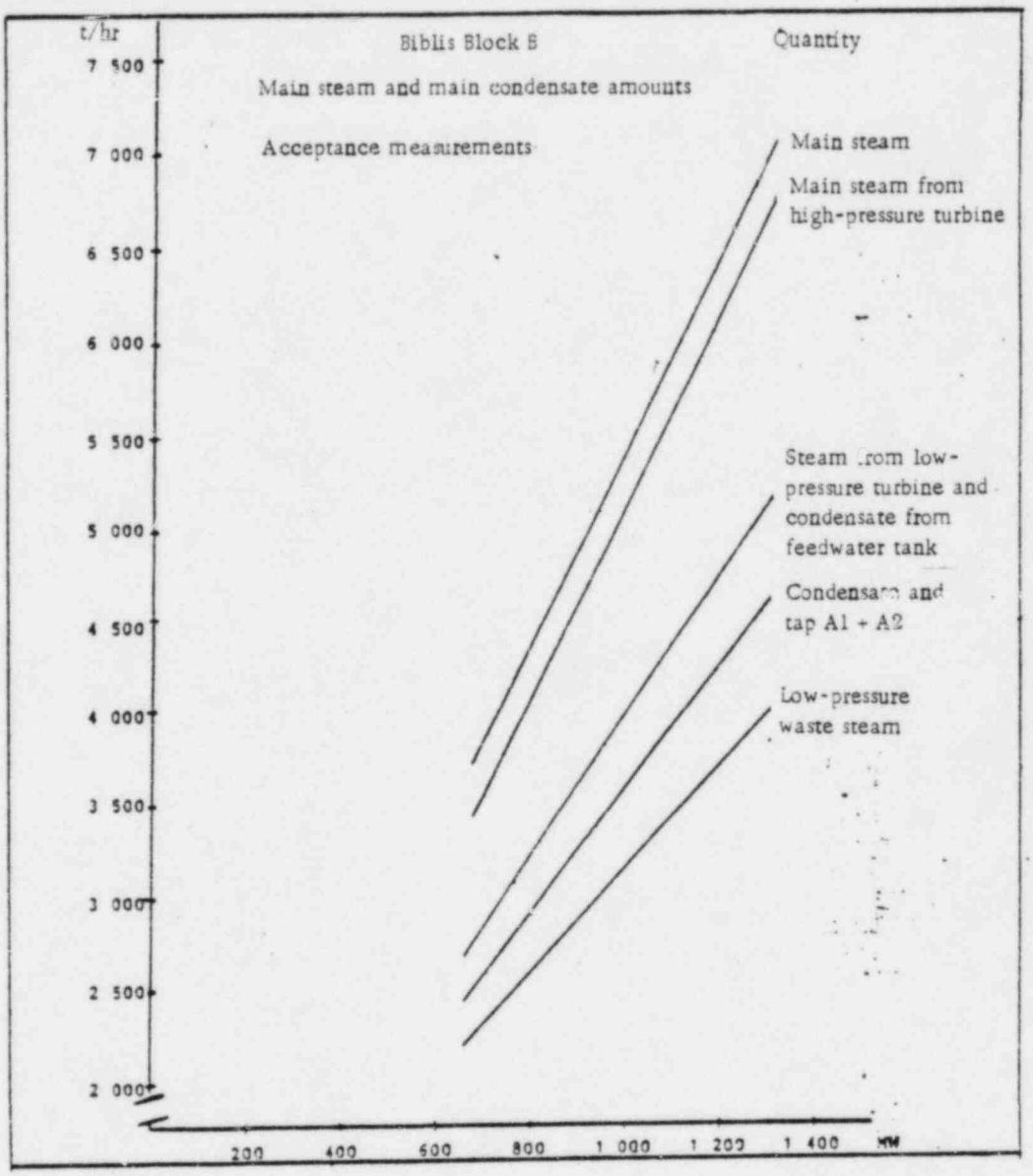


Figure 2.6.2. Acceptance Measurements of Biblis B.

2.6.3. Steam Emissions Through the Power House Roof

2.6.3.1. Steam Emissions at Constant Load

The amount leaked by the water-steam loop is made up for by the demineralized water system. The demineralized water is manually injected into the hot well from the control room of the unit via a control valve on the basis of the level in the feedwater tank.

The amount of water fed into the hot well is determined by means of an oriface gauge and corresponds to the amount leaked from the water-steam circuit.

-- In the Obrigheim NPP (gross power approximately 345 MWe), the mean amount of demineralized water reinjected varied between 1.4 t/hr and 9.3 t/hr, in this process reinjected amounts of 1-3 t/hr were most frequent and amounts of 3-4 t/hr were the next most frequent [4].

-- In the Stade NPP (gross power 662 MWe), the mean amount of demineralized water reinjected varied between 3.4 t/hr and 9.3 t/hr; reinjection amounts of 4-5 t/hr were the most frequent, and amounts of 9-10 t/hr and 5-7 t/hr were the next frequent [4].

Stade, which has approximately twice as much power as Obrigheim, also has approximately twice as much leakage from the water-steam circuit.

If we now consider the Biblis B NPP which, with a generator terminal power of 1,300 MW, has approximately double the power of Stade, on the basis of the above-depicted tendency approximately twice as much leakage from the water-steam circuit can be expected [from Biblis] as from Stade. This means that the mean amount of demineralized water reinjected was between 6 t/hr and 18 t/hr, while reinjection quantities of 8-10 t/hr were most common.

These values are quite consistent with the statements of the operator, who indicates a mean reinjection quantity of 8 t/hr for December 1979 for Biblis B [5] (see footnote on next page).

From the above-cited statements, a mean amount of demineralized water of approximately 10 t/hr reinjected into the water-steam circuit must be assumed for a 1,300 MW_{el} NPP. How is this quantity divided into steam leaks and water leaks?

If the water leakage were known, perhaps by exact measurement of the building draining, it would be possible to determine the steam leaks therefrom since the total amount of demineralized water feed is known. For various reasons, the amount of water fed is, however, not known precisely. Let us mention here only two reasons: in many cases, the water which is fed is measured inductively. In the case of leaky check valves, the water is recorded several times, thus greatly distorting the measured values. Another problem is condenser leaks through which an additional unknown quantity of water gets into the feedwater circuit.

Estimates concerning steam leaks in the high-pressure section and concerning the feedwater degasification system led to the following results, which are also verified by the management of Biblis as being realistic [6]:

-- Steam leaks in the high-pressure section of approximately 3.6 t/hr (essentially the high-pressure drain flash tank with a stand pipe to the power house roof, leaks at stuffing boxes, sliding valves, etc.).

-- Steam leaks through the feedwater degasification system of approximately 2 t/hr (in nuclear power facilities up to the level of Biblis B, the feedwater degasification system has a stand pipe with an outlet via the power house roof).

In the case of leaks which are limited in time, these values can also be considerably higher. At this point let us refer to a publication by F. J. Spalthoff [7] in which steam leaks in the high-pressure section of the turbine are described.

In the parameter study, these values are generally used for the computations. In order to obtain an overview of the effects of the amounts of steam leaked on the overall emission, these values were also varied as follows.

Amounts of steam leaked in the high-pressure section 2-10 t/hr.

Amounts of steam leaked through the feedwater degasification system 0-6 t/hr.

In the parameter study for Unit 5 - extreme calculation, the steam leakage rate in the high-pressure section was also varied from 1 to 100 t/hr.

If the steam leakage of the feedwater degasification system is set equal to zero, this will make it possible to simulate newer plants than Biblis B (for instance, the Grohnde NPP) in which the exhaust steam of the feedwater is fed to the evaporator overhead condenser and is not released through a stand pipe via the power house roof.

The water leaks into the sump are summarily assumed to be 4.5 t/hr in most of the calculations.

- All other operators of NPPs in the FRG have either failed to respond at all to corresponding requests or have referred the requestors to other quarters, usually the KWU. In all cases the end result was that no data were made available. This refusal to provide information must be viewed in light of the fact that in the public and even in the government it is stated again and again that science and the nuclear industry in particular have an obligation to provide information to the citizens.

2.6.4. Bibliography for 2.6

1. "Report of the Rhine-Westphalian Utility, Biblis Management", 6/15/81.

2. KWU (publisher), "Das Kernkraftwerk Biblis des RWE" [The Biblis NPP of the RWE], Von Mandel, H. et al., from the series: "Pattern Plants in Energy Management", Energiewirtschaft und Technische Verlagsgesellschaft m.b.H., Graefelfing-Munich.

3. Atomwirtschaft-Atomtechnik, special reprint from Vol. 8/9, pp. 428-429, Aug./Sept. 1974.

4. Kraftwerkunion (KWU) (publisher): "Amounts of Demineralized Water Fed into the Secondary Loop", RD 12/4 000/202000/Bu, 2/24/75.

5. Report from the management of Biblis dated 1/9/80: "Documents on Additional Quantities of Demineralized Water Fed into the Secondary Section".

6. Personal report from the Biblis management during a visit on 6/12/81.

7. Spalthoff, F. J., "Initial Inspections of Unit A of the Biblis NPP", AED conf., 1976, pp. 183-200.

2.7. Information Procurement

Safety reports, documents and the results of plant-specific measurements, calibration curves of measurement instruments (for instance, N-16 detectors) are some of the significant data required for the scientific processing of the topic described here. In many cases, these data are not available through the data banks and technical information centers since they are the subject of proprietary reservations and protective covenants.

Obtaining such data through official channels proved to be impossible for this study and can only be regarded as a hindrance to scientific investigation in terms of both time and content.

As an example, let us at this point mention the attempts made to obtain the safety report for Biblis B:

-- On 8/15/79, information from the Karlsruhe Technical Information Center: safety reports (SR) are confidential in the FRG, i.e., not open to the public. (In the U.S.A. it is different; American safety reports are thus available.) We were asked to contact the GRS for the safety reports for German NPPs.

-- On 9/19/79, information from the GRS: the GRS does have the safety reports but is not allowed to provide them to others. We were asked to contact the Federal Minister of the Interior [BMI].

-- On 9/26/79, letter to the BMI requesting that the safety report for Biblis B be made available.

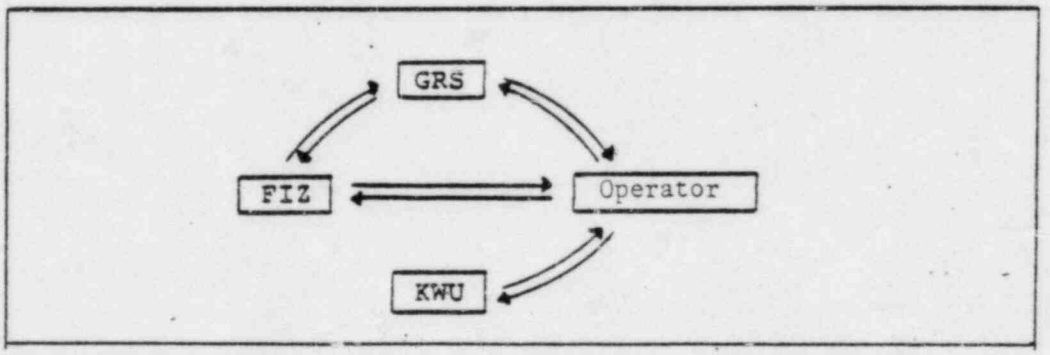
-- On 11/3/79, the BMI referred us to the proprietor of the safety report, the Rhine-Westphalian Utility, Inc. in Essen.

-- On 11/28/79, a visit was made to the Biblis NPP. The plant management promised us that we would receive the safety reports.

-- On 1/7/80, visit to the Biblis NPP. The management informed us that we could not obtain the safety reports since the upper-level management in Essen did not want to allow us to do so.

We had similar experiences in the procurement of other unpublished data:

Thus, the Reactor Safety Company referred us to the Technical Information Centers and the power plant managements. The power plant managements either did not respond to the requests at all or referred us in most cases to the KWU. (The only exception was the Biblis NPP!) The KWU, in turn, responded to our request that the managements made the data in question available to them for personal use only and that the KWU was not allowed to forward such data. Each institution referred to other institutions without providing any data. An analysis of a portion of the correspondence shows:



GRS - Reactor Safety Company

FIZ - Technical Information Center

KWU - Power Plant Union

Figure 2.7. Requests for Unpublished Data were not Responded to By the Above-Indicated Institutions, But Rather were Handled by Referring to Other "More Cognizant" Institutions.

It should be mentioned in particular that even the Kahl Test NPP would not provide any in-house data even though its construction was financed almost exclusively from public funds. In the meantime 80% of the ownership of it has been shifted to the Rhine-Westphalian Utility (RWE) and 20% to the Bavarian Works, and the previous public backers are, allegedly, unable to elicit data from them.*

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*Even the Federal Ministry for Research and Technology was unable to procure the requested documents (for instance, the safety report for Biblis B) for the IFEU although the Federal Ministry of the Interior and the GRS have these documents.

The fact that it was still possible to successfully complete this work is due to many anonymous employees of the above-listed institutions who ignored the official "information embargo" of the companies which employ them. Their willingness to help came mostly from dissatisfaction with, in particular, the the information policies of their employers.

3. Computer Program

3.1. Program Structure

The SEKEM4 program is used to simulate the activity distribution in the secondary loop in the event of SG leaks and to quantitatively determine activity emissions from the secondary loops of light-water pressure reactors.
To take into account time-dependent processes, for instance concentration spikes in the primary loop or a change in the power of the reactor, SEKEM4 solves the following heterogeneous linear differential equation:

$$\frac{d}{dt}\vec{A}(t) = TM(t) \cdot \vec{A}(t) + \vec{Q}(t) \qquad (1)$$

TM(t) - Time-dependent transport matrix,

$\vec{A}(t)$ - Activity distribution in the secondary loop,

$\vec{Q}(t)$ - Source intensities [may also be "source thicknesses", "source strengths"].

The emissions are calculated from the activity variation $\vec{A}(t)$.

A fourth-order Runge-Kutta procedure for stiff differential equations [1] is used to solve linear differential equations ([1].

The linear equation system

$$TM \cdot \vec{A}_S + \vec{Q}_S = 0 \qquad (2)$$

results as a special case from linear differential equation (1) on the basis of the following assumptions.

$$\lim_{t\to\infty}\left[\frac{d}{dt}\vec{A}(t)\right] = 0, \quad \lim_{t\to\infty}\left[\frac{d}{dt}\vec{Q}(t)\right] = 0. \qquad (2a)$$

$$\vec{A}_S = -TM^{-1} \cdot \vec{Q}_S$$

$\vec{A}_S$ - Steady-state activity distribution,

$\vec{Q}_S$ - Steady-state source intensity.

Equation (2a) represents the activity distribution in equilibrium.

The SEKEM4 program was written on the IBM/370-168 system of the computer center of the University of Heidelberg. The WATFIV-Compiler was used to develop and test the program.

The final version of SEKEM4 is written in standard FORTRAN IV.

The superior structure of the program is shown in Figure 3.1.

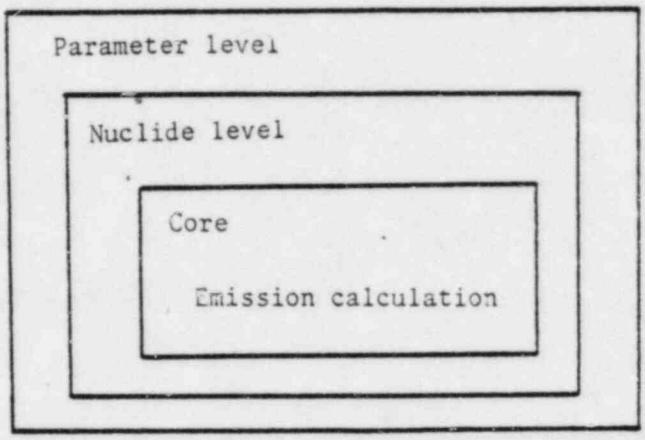

Figure 3.1. Diagram of the Program Structure of SEKEM4 for Calculating Secondary Loop Emissions from Light-Water Pressure Reactors.

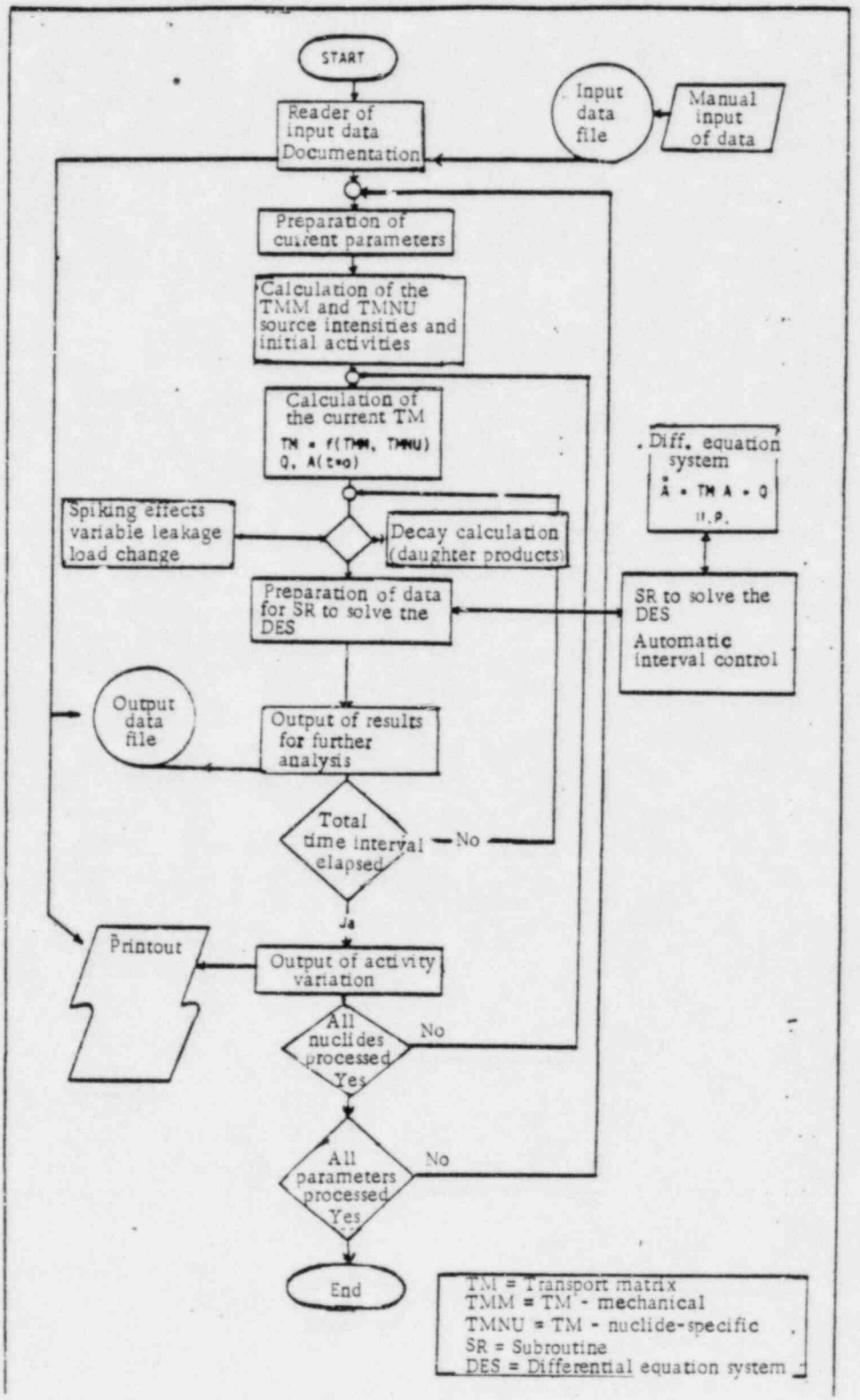

Figure 3.2. Schematic Block Diagram of the SEKEM4 Computer Program.

The core of the program calculates the activity distribution and the emission of a certain nuclide with a preset parameter combination.

On the nuclide level, the program core is called for each nuclide.

In the successive servicing of the nuclide table, entire decay chains (several daughter products) in the secondary loop can be taken into account. This is used in particular to examine the activity distribution of decay products of fast-decaying noble gases.

Due to considerations of storage space and computer time, the nuclide table used in one computer run was restricted to 30 nuclides.

If the program has serviced the nuclide table for a certain set of parameters, it returns to the parameter level and calculates a new parameter combination.

Figure 3.2 shows a schematic block diagram of the SEKEM4 computer program.

The parameters to be varied can be selected as desired. This applies to the sequence, type, range of variation and type of variation. Up to 100 parameters can be processed per program run.

SEKEM4 consists of approximately 1,700 FORTRAN instructions and occupies approximately 600 KB [kilobytes, uncertain] on the IBM 370.

The following rule of thumb has been adopted for estimating the transit time to be expected T_{tot}:

$$T_{tot} = 0{,}01\ j \prod_{l=1}^{n} m_l\ [\text{sec}] \qquad (3)$$

T_{tot} - Transit time to be expected,

j - Number of nuclide,

n - Number of parameter,

m_l - Number of the parameter values of the l-th parameter.

This transit time estimation applies only to emission calculations in system equilibrium.

The computer time required for solving the differential equation in the case of time-dependent processes cannot be estimated in advance since it depends on the gradients of the operands.

The time required for a computer run is greatly dependent on the number of parameters and the number of parameter values per parameter. The extent of the outputs of SEKEM4 is also heavily dependent on the number of parameter values. In order to keep the computer time and extent of the output within reasonable bounds, it is advisable to limit the parameter combinations.

The developed program packet includes the programs SORTSEKEM and PLOT. SEKEM transfers the emission data to the SORTSEKEM program. This program takes from

the output file the emission values which correspond to a preset parameter combination. The plot program displays the sorted-out data on graph output devices.

Figure 3.3 shows the data flow diagram of SEKEM4 in schematic form. In simple application cases, the temporary auxiliary data files are not necessary.

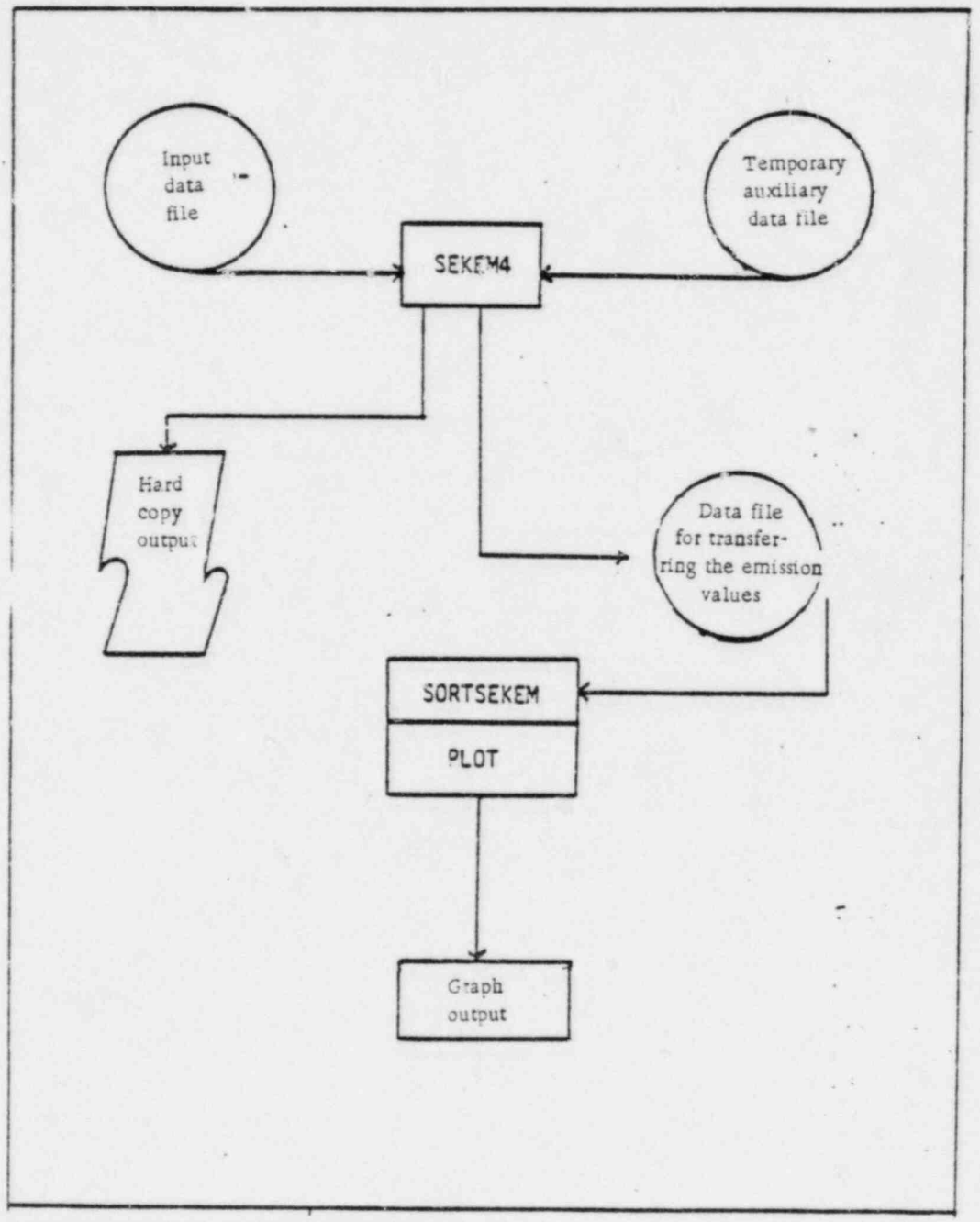

Figure 3.3. Simplified Data Flow Diagram of SEKEM4 for Calculating Secondary Loop Emissions from Light-Water Pressure Reactors.

3.2. Program Description

SEKEM4 simulates the transport of activity in the secondary loop as well as the emissions from the secondary loop of a PWR. The program contains a standard data record for the secondary loop. This record corresponds to that of the Biblis B NPP.

This data record is incorporated into the transport matrix TM (compare equation (1)). In principle, it is possible to modify the data record from the outside in order to use SEKEM4 for simulating other secondary loops as well.

The main program of SEKEM4 shows the program structure (compare listing 3.1). The core consists of the calls for the subroutines BALANC and PROCEE. The NEWTM subroutine works on the nuclide level. After leaving the nuclide loop, a new parameter combination is calculated in the subroutines PARAME and TRANSL.

The INPUT subroutine reads the control data in and, if necessary, modifies the data record to set the secondary loop of a certain NPP.

The calculation of the activity distribution in the secondary loop and the resulting emissions is done in the following steps:

I. Loading of the current parameter values by the TRANSL and PARAME subroutines.

PARAME processes a general parameter list in accordance with the stored initial values, step widths and final values.

The possible parameters are divided into the following groups:

-- Reactor power

-- Nuclide-specific activities in the primary loop

-- SG leakage

-- Mass contents

-- Mass flows

-- Transport times

-- Nuclide-specific α values (phase distribution factors) in the high-pressure and low-pressure sections

-- Residual moistures

-- Decontamination factors in the blowdown demineralization system.

TRANSL alters the data record in accordance with the parameter list.

```
                     SYSTEM/370 FORTRAN H EXTENDED (ENHANCED)
REQUESTED OPTIONS: AD(DBLPAD4),NAME=SEKEM4
OPTIONS IN EFFECT: NAME(SEKEM4) OPTIMIZE(3) LINECOUNT(66) SIZE(MAX) AUTODBL(DBLPAD4)
                   SOURCE EBCDIC NOLIST NODECK OBJECT MAP NOFORMAT GOSTMT NOXREF ALC NOANSF NOTERM IBM FLAG(I)

              C ****************************************************************
              C *******************                     *******************
              C *******************                     *******************
              C *******************        SEKEM4       *******************
              C *******************     MAINPROGRAMM    *******************
              C *******************                     *******************
              C *******************                     *******************
              C ****************************************************************
              C ****************************************************************
              C **     IFEU  HEIDELBERG                          20.10.1981     **
              C
              C
              C
ISN 0002            INTEGER*4 ALLNUK,ALLPAR,IOCV(10)
              C
              C             COMMON
ISN 0003            COMMON /CONTR1/IPROCE,ALLNUK,ALLPAR,IRADIO,ISPIKE,IOCV
              C
              C
              C
              C                  START
              C
              C
ISN 0004            CALL INPUT
ISN 0005          1 CALL TRANSL
              C
              C                  PARAMETERLOOP
              C
              C
              C                  NUKLIDLOOP
              C
              C
ISN 0006          2 CALL MENTH
ISN 0007            CALL BALANC
              C
ISN 0008            IF(IPROCE .NE. 0)   CALL PROCEL
              C
ISN 0010            IF(ALLNUK .EQ. 0)   GOTO 2
              C
ISN 0012            CALL PARAME
ISN 0013            IF(ALLPAR .EQ. 0)   GOTO 1
              C
ISN 0015            CALL MOVEPI(15,6)
ISN 0016            CALL MOVEPI(14,6)
              C
ISN 0017            STOP
ISN 0018            END
```


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Listing 3.1. Main Program of SEKEM4.

```
SYSTEM/370 FORTRAN H EXTENDED (ENHANCED)

                                        /  SEKEM4 /     SIZE OF PROGRAM 000198 HEXADECIMAL BYTES

  NAME    TAG   TYPE  ADD.       NAME    TAG   TYPE  ADD.       NAME    TAG   TYPE  ADD.       NAME    TAG   TYPE  ADD.
  IOCF      CP  I*4     48      INPUT SF  XF          000000    NEWTA SF  XF          000000    ALLMUX    CP  I*4   000008
ALLPAR      CP  I*4   000010    BALANC SF  XF          000030   IBCOM#     XF  I*4   000000    IPROCE    CP  I*4   000000
IRADIO      CP  I*4     58      ISPIKE      CP  I*4     48      MOVEFI SF  XF          000000    PARAME SF  XF          000000
PROCEE SF  XF           000030  TRANSC SF  XF          000030

                                        *****  COMMON INFORMATION  *****

     NAME OF COMMON BLOCK  *CONTRI*   SIZE OF BLOCK    000078 HEXADECIMAL BYTES

   VAR. NAME  TYPE  REL. ADDR.   VAR. NAME  TYPE  REL. ADDR.   VAR. NAME  TYPE  REL. ADDR.   VAR. NAME  TYPE  REL. ADDR.
     IPROCE   I*4    000000       ALLMUX   I*4    000008       ALLPAR   I*4    000010       IRADIO   I*4    000018 RB
     ISPIKE   I*4    000020 RB      IOCF   I*4    000028 RB

 SOURCE STATEMENT LABELS
   LABEL  ISN    ADDR          LABEL  ISN   ADDR          LABEL  ISN   ADDR          LABEL  ISN   ADDR
       1    5  0000EC              2    6  0000F8
 COMPILER GENERATED LABELS
   LABEL  ISN    ADDR          LABEL  ISN   ADDR          LABEL  ISN   ADDR          LABEL  ISN   ADDR
  100000    1  00000C       100001    9  000114       100002   10  000126       100003   12  000136
  100004   15  000146
*OPTIONS IN EFFECT*NAME(SEKEM4) OPTIMIZE(3) LINECOUNT(66) SIZE(MAX) AUTODBL(DBLPAD4)
*OPTIONS IN EFFECT*SOURCE EBCDIC NOLIST NODECK OBJECT MAP NOFORMAT OFFSET NOXREF ALC NOANSF NOTERM IBM FLAG(I)
*STATISTICS*     SOURCE STATEMENTS =       17, PROGRAM SIZE =      390, SUBPROGRAM NAME =SEKEM4
*STATISTICS*  NO  DIAGNOSTICS GENERATED
****** END OF COMPILATION ******                                  172K BYTES OF CORE NOT USED
```


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Listing 3.1 (Continued). Main Program of SEKEM4.

II. The NEWTM subroutine calculates the transport matrix with the aid of the current data record and the corresponding nuclide data.

The point-specific data summarized in Section 2.6 are incorporated into the calculation of the transport matrix as follows:

$$TM_{ij} = \frac{m_{ij} \cdot (\alpha_k \cdot f_{ij} + 1 - f_{ij})}{M_i \cdot (\alpha_k \cdot f_{ii} + 1 - f_{ii})} \exp(-\lambda_k \cdot TRT_{ij}) \quad \text{if } i \neq j$$

$$TM_{ii} = -\lambda_k - \sum_{j=1}^{N} TM_{ij}$$

m_{ij} - Mass flow from point i to point j

M_i - Mass content of point i

f_{ij} - Moisture in mass flow m_{ij}

f_{ii} - Moisture at point i

TRT_{ij} - Transport time from point i to point j

α_k - Solubility in steam for the k-th nuclide

λ_k - Decay constant for the k-th nuclide.

In the BALANC subroutine the activity distribution in steady-state equilibrium is calculated according to equation (2a).

From these activities, the sum of the emissions in equilibrium is calculated as follows:

$$e_{roof} = \sum_{i=1}^{N} A_i \, m_i \, M_i^{-1}$$

M_i - Mass content of the i-th point

m_i - Steam emission of the i-th point

A_i - Activity of the i-th point.

The sum extends over all points under consideration. For noble gases, the pathway of condenser exhaust-stack emission is also calculated. This activity or emission calculation can also be performed for different primary concentrations. From the calculated activity distribution in equilibrium result possible initial values for the solution algorithm to differential equation (1). The NEWTM subroutine has loaded the auxiliary matrix ZTM to determine the source intensities for daughter products.

For the source intensity vector, the following is true:

$$\vec{Q}(t)_T = ZTM(t) \cdot \vec{A}(t)_M + \vec{Q}(t)_{\text{Primary}} \qquad (4)$$

$\vec{Q}(t)_T$ - Total source intensity for the daughter product

$\vec{Q}(t)_{primary}$ - Primary source intensity for the daughter product

$\vec{A}(t)_M$ - Activity distribution of the precursor

ZTM - Decay transport matrix.

III. The PROCEE subroutine solves differential equation (1).

The total emissions through the roof $A(t)_{roof}$ are calculated from the values of A(t). Before A(t) is calculated, the power of the reactor, the SG leakage and the primary loop concentration at time t are determined. This yields the time-dependent transport matrices TM(t), ZTM(t) and the source intensities Q(t) in (1) and (4). In examining the decay chains, the source intensity $Q(t)_T$ for the daughter product is calculated from the activity distribution for the precursor $A(t)_M$.

The SORTSEKEM and PLOT programs, which prepare the results and display them in graphic form, are especially tailored to the output devices at the computer center of the University of Heidelberg. Their general applicability is thus not guaranteed.

3.3. Examples of Input and Output Data Files as well as of Plot

A typical input data file is shown in listing 3.2.

```
STATUS INFORMATION
IPROCT= 0  ISPIKE= 0  IRADIO= 0                                  Nuclide NR = Nuclide no.
IOCV =  1  0  0  0  0  0  0  1  0  0
DELCK= 5.56D-06  PGWIB=  1.0
DELT= 6.05D+05  III =    0


RP( 3,13)= 1.000-03  TET( 3,13)= 0.0       F( 3,13)= 5.000-03
ALPHAR   = 1.000+00                        ALPHAL   = 1.000+00 DFALL= 1.000+00
LAB( 1)= 1.000+00  PRLRC( 1) = 1.000+00 NAME NUKLID NR.: 1   L= 1
DFABS( 1)= 1.000+00  PVF( 1, 1)= 1.000+03
DFABS( 1)= 1.000+00  PVF( 5, 1)= 1.000+05
LAB( 2)= 1.000-06  PRLRC( 2) = 1.000+00 NAME NUKLID NR.: 2   L= 6
DFABS( 2)= 1.000+00  PVF( 1, 2)= 1.000+03
DFABS( 2)= 1.000+00  PVF( 5, 2)= 1.000+05
LAB( 3)= 1.000-12  PRLRC( 3) = 1.000+00 NAME NUKLID NR.: 3   L=12
DFABS( 3)= 1.000+00  PVF( 1, 3)= 1.000+03
DFABS( 3)= 1.000+00  PVF( 5, 3)= 1.000+05
IP=  1 ISTART= 0.0       ISTEP= 1.11D-03 IEND= 6.70D-03  ICODE= 2011001  VARNAME:SP
IP=  2 ISTART= 1.000+00  ISTEP= 1.000+01 IEND= 1.10D+04  ICODE=18030002  VARNAME:DFAL
IP=  3 ISTART= 0.0       ISTEP= 2.78D-04 IEND= 1.70D-03  ICODE= 2091303  VARNAME:SP
```

Listing 3.2. Typical Data Input File for SEKEM4.

Listing 3.3 contains an excerpt of the corresponding output data.

Power house roof - emissions

SP T/S	DFAC	SP T/S	[illegible]	[illegible]	[illegible]	Total
0.0	1.000+00	0.0	8.420-16	1.070-07	2.470-06	2.580-06
1.110-03	1.000+00	0.0	8.420-16	1.040-07	1.870-06	1.970-06
2.220-03	1.000+00	0.0	8.420-16	1.010-07	1.310-06	1.410-06
3.330-03	1.000+00	0.0	8.420-16	9.900-08	1.250-06	1.350-06
4.440-03	1.000+00	0.0	8.420-16	9.670-08	1.190-06	1.290-06
5.560-03	1.000+00	0.0	8.420-16	9.450-08	1.130-06	1.230-06
6.670-03	1.000+00	0.0	8.420-16	9.240-08	8.500-07	9.430-07
0.0	1.000+01	0.0	8.420-16	1.070-07	2.470-06	2.580-06
1.110-03	1.000+01	0.0	8.420-16	9.550-09	1.350-08	2.200-08
2.220-03	1.000+01	0.0	8.420-16	5.100-09	5.240-09	1.020-08
3.330-03	1.000+01	0.0	8.420-16	3.100-09	3.500-09	6.880-09
4.440-03	1.000+01	0.0	8.420-16	2.560-09	2.620-09	5.180-09
5.560-03	1.000+01	0.0	8.420-16	2.060-09	2.100-09	4.150-09
6.670-03	1.000+01	0.0	8.420-16	1.710-09	1.750-09	3.470-09
0.0	1.000+02	0.0	8.420-16	1.070-07	2.470-06	2.580-06
1.110-03	1.000+02	0.0	8.420-16	3.130-09	9.350-09	1.910-09
2.220-03	1.000+02	0.0	8.420-16	4.770-09	4.980-09	9.750-09
3.330-03	1.000+02	0.0	8.420-16	3.240-09	3.320-09	6.550-09
4.440-03	1.000+02	0.0	8.420-16	2.440-09	2.490-09	4.930-09
5.560-03	1.000+02	0.0	8.420-16	1.960-09	2.000-09	3.950-09
6.670-03	1.000+02	0.0	8.420-16	1.640-09	1.660-09	3.300-09
0.0	1.000+03	0.0	8.420-16	1.070-07	2.470-06	2.580-06
1.110-03	1.000+03	0.0	8.420-16	9.390-09	9.900-09	1.930-08
2.220-03	1.000+03	0.0	8.420-16	4.750-09	4.960-09	9.710-09
3.330-03	1.000+03	0.0	8.420-16	3.210-09	3.310-09	6.520-09
4.440-03	1.000+03	0.0	8.420-16	2.430-09	2.480-09	4.910-09
5.560-03	1.000+03	0.0	8.420-16	1.950-09	1.990-09	3.940-09
6.670-03	1.000+03	0.0	8.420-16	1.630-09	1.660-09	3.290-09
0.0	1.000+04	0.0	8.420-16	1.070-07	2.470-06	2.580-06
1.110-03	1.000+04	0.0	8.420-16	9.060-09	9.490-09	1.900-08
2.220-03	1.000+04	0.0	8.420-16	4.740-09	4.960-09	9.700-09
3.330-03	1.000+04	0.0	8.420-16	3.210-09	3.310-09	6.520-09
4.440-03	1.000+04	0.0	8.420-16	2.430-09	2.480-09	4.910-09
5.560-03	1.000+04	0.0	8.420-16	1.950-09	1.990-09	3.940-09
6.670-03	1.000+04	0.0	8.420-16	1.630-09	1.650-09	3.280-09
0.0	1.000+00	2.780-04	8.420-16	1.210-07	2.650-06	2.770-06
1.110-03	1.000+00	2.780-04	8.450-16	1.240-07	2.120-06	2.250-06
2.220-03	1.000+00	2.770-04	8.530-16	1.310-07	1.810-06	1.940-06
3.330-03	1.000+00	2.730-04	8.640-16	1.260-07	1.610-06	1.740-06
4.440-03	1.000+00	2.730-04	8.790-16	1.400-07	1.450-06	1.590-06
5.560-03	1.000+00	2.740-04	8.970-16	1.440-07	1.340-06	1.490-06
6.670-03	1.000+00	2.730-04	9.180-16	1.440-07	1.260-06	1.400-06
0.0	1.000+01	2.780-04	8.420-16	1.210-07	2.650-06	2.770-06
1.110-03	1.000+01	2.740-04	8.450-16	1.130-08	1.240-08	2.360-08
2.220-03	1.000+01	2.780-04	8.520-16	6.110-09	6.420-09	1.250-08
3.330-03	1.000+01	2.780-04	8.620-16	4.290-09	4.430-09	8.720-09
4.440-03	1.000+01	2.790-04	8.740-16	3.360-09	3.440-09	6.790-09
5.560-03	1.000+01	2.780-04	8.880-16	2.780-09	2.840-09	5.620-09
6.670-03	1.000+01	2.780-04	9.040-16	2.400-09	2.440-09	4.840-09
0.0	1.000+02	2.780-04	8.420-16	1.210-07	2.650-06	2.770-06
1.110-03	1.000+02	2.730-04	8.450-16	1.040-08	1.170-08	2.210-08
2.220-03	1.000+02	2.790-04	8.520-16	5.620-09	6.080-09	1.190-08
3.330-03	1.000+02	2.730-04	8.610-16	4.070-09	4.190-09	8.270-09
4.440-03	1.000+02	2.780-04	8.740-16	3.160-09	3.250-09	6.420-09
5.560-03	1.000+02	2.790-04	8.890-16	2.620-09	2.680-09	5.310-09
6.670-03	1.000+02	2.780-04	9.060-16	2.270-09	2.300-09	4.570-09
0.0	1.000+03	2.780-04	8.420-16	1.210-07	2.650-06	2.770-06
1.110-03	1.000+03	2.790-04	8.450-16	1.070-08	1.170-08	2.280-08

Listing 3.3. Output Data for the Input Values in Listing 3.2.

Figure 3.4 shows a plot resulting from listing 3.3.

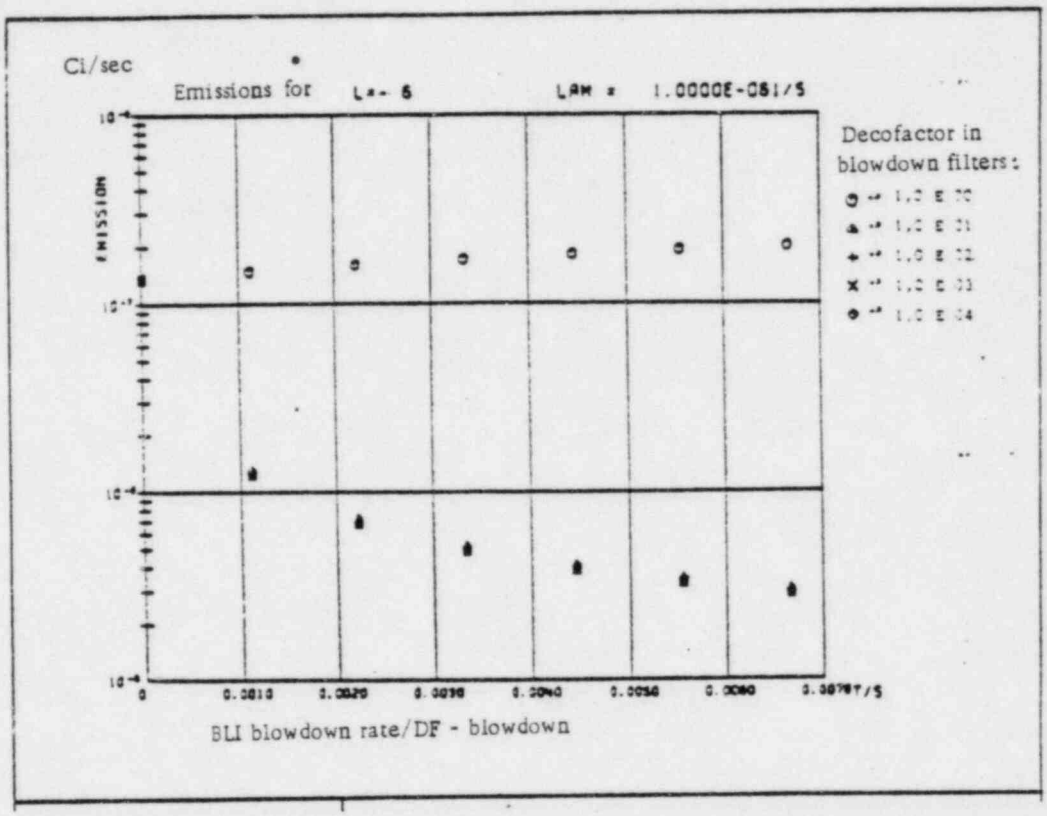


Figure 3.4. Power House Emissions in Ci/sec as a Function of the Blowdown Rate in t/sec for a Nuclide with $\lambda = 10^{-6}$ sec^{-1} (compare Iodine 131).

In this calculation, the following were selected as constant parameters: SG leakage in the defective SG (20 kg/hr), primary loop activity (1 Ci/t), residual moisture in the SG (0.25%) and steam emissions from the high-pressure section (3.6 t/hr).

The variable parameters were varied within the following limits:

	Initial value	Step width	End value	Unit
Blowdown rate	0	4	24	t/hr
Decofactor in blowdown demineralization system	1	10	10^4	
Steam emissions from feedwater degasification system	0	1	6	t/hr

In the plot shown (Figure 3.4), the total emission through the power house roof, in Ci/sec, is plotted on the ordinate. The blowdown rate in t/sec is plotted on the abscissa. The decontamination factors in the filter of the blowdown demineralization system were selected as parameters.

3.4. Bibliography for 3

1. Stoer, J. and R. Bulirsch, "Einfuehrung in die Numerische Mathematik I and II" [Introduction to Numerical Mathematics I and II], Berlin, Heidelberg, New York, 1978.

2. Kaps, P. and P. Rentrop, "Generalized Runge-Kutta Methods of Order Four with Stepsize Control for Stiff Ordinary Differential Equations", NUMER. MATH., Vol. 33, pp. 55-68, 1979.

3. Selder, H., "Einfuehrung in die Numerische Mathematik fuer Ingenieure" [Introduction to Numerical Mathematics for Engineers], Munich, Vienna, 1979.

4. Sensitivity Analysis (Parameter Study)

- Introduction

The purpose of a parameter study is to describe with adequate precision a preset system by analyzing its specification factors themselves and their functional dependencies. In the present studies, the system consists of the secondary loop components which can contribute to the emission of radioactivity. The specification factors of the secondary loop system with respect to emission are the so-called emission parameters, the interaction of which creates a certain emission. The emission E is thus a function, the n emission parameter p_i applies in formal terms [Translator's note: the grammar of this sentence is somewhat confusing and may be faulty; it is possible that the sentence should read something like "The emission E is thus a function to which n emission parameters p_i are applicable in formal terms"]:

$$E = f_E (p_1, p_2, \cdots p_n),$$

where individual parameters are, in turn, a function of a portion of the other parameters:

$$p_i = f_i (p_1, \cdots, p_{i-1}, p_{i+1}, \cdots p_k) \text{ with } 1, k \leq n.$$

Since the functions f_E and f_i are unknown, within the framework of a sensitivity analysis it is determined which of the n emission parameters strongly affect the emission, and which affect it only slightly. This procedure consists in repeatedly calculating the emission for different parameter constellations which are to be established in a directed manner.

In view of the high degree of complexity of the secondary loop system and the degree of differentiation of the selected system model (compare Section 2.1, model of the secondary loop), it is a good idea to carry out a parameter study or the sensitivity analysis inherent in it for the following reasons:

1. The correctness of the computer program as well as the logical consistency of the model assumptions summarized in the simulation model can be checked by calculated variation of the sets of parameters.

2. By varying the parameters between the minimum and maximum limits, the hypothetical or theoretical nuclide-specific emission potential is calculated. The presentation of the emission is thus complete within the framework of the pre-establishment of the parameters.

3. For any operating state, on the other hand, the corresponding parameter constellation and thus the maximum emission are to be obtained.

4. Since the theoretical emission spectrum is being calculated, the assessment of the probable emission is to be carried out in a clearly delineated second step, in which plant-specific data must be taken as the point of departure in each case.

5. Finally, an evaluation of the sensitivity analysis allows calculated optimization of the operating parameters with a view to minimum emission of radionuclides.

In the following sections remarks are first presented regarding the method of the parameter study or sensitivity analysis, and then the selected calculations of the quantitative analysis are presented, evaluated and finally summarized.

4.1. Remarks Regarding the Method

The secondary loop model used in this study results from an optimization of the competing preset goals of structural accuracy and degree of detail as opposed to program expense and computation costs. The same applies to the selection of the variation parameters. The following 11 parameters were determined as important variation parameters:

1). Primary loop activity A_n in [Ci/t] of nuclide n

2). SG leakage L in [t/hr]

3). Decay constant λ_n in [sec^{-1}] of nuclide n

4). Steam emission rate D_H from the high-pressure section in [t/hr]

5). Steam emission rate D_L from the low-pressure section (feedwater degasification system) in [t/hr]

6). Residual moisture f^*_{SG} in steam generator SG in [%]

7). Blowdown rate ABS per SG in [t/hr]

8). Residual moisture f^*_{ABS} in the feedwater heating steam from the blowdown demineralization flash tank in [%]

9). Decofactor DF_{ABS} in the blowdown demineralization system

10). Phase distribution factor α^*_H in the high-pressure section

11). Phase distribution factor α^*_L in the low-pressure section or the ratio α_L/α_H.

The operating parameters of mass flows and transport times which directly affect emission were not included in the detailed examination of the emission parameters. As explained in greater detail below, this is taken into account by distinguishing the cases of normal operation (100% power) and load change with activity spikes.

Both the computer expense and the material itself make it impossible to determine the emissions from the secondary loop for every possible combination of parameters. Thus, for instance, there is no sense in calculating emissions for variable residual moistures f_{ABS} in the feedwater heating steam

---

*The relationship between the residual moisture f, the phase distribution factor α and the decofactor is explained in detail in Section 2.2, Mathematical Model of the Phase Distribution or Decontamination Factors, as well as Section 2.5, Decontamination Factors.

from the blowdown demineralization flash tank (compare also Section 2.1, model of the secondary loop of a light-water reactor) if at the same time the blowdown rate is set at ABS = 0. The same applies, for instance, to emission calculations of noble gases ($\alpha_H = 10^{-4}$) in the event of a variable residual moisture f_{SG} in the SG since the noble gases are transported almost exclusively with the steam phase.

For a systematic analysis of the parameters, six blocks were created which are defined from their correspondence to specific functions of a certain system part of the secondary loop or which correspond to special operating conditions.

The blocks are:

1. Blowdown demineralization ABS

2. High-pressure section

3. Low-pressure section

4. Standard calculation

5. Extreme calculation

6. Spike calculation and load change.

Two sets of parameters were formed for each block. The first set contains parameters which are particularly relevant to the corresponding block and are therefore to be varied, while the second set contains the parameters which are set at a constant value. The parameter values of the parameters in the second set correspond, as a rule, to the values of the so-called standard calculation (block 4). For the sake of illustration, let us briefly describe block 1 here.

For studying the possible effects of blowdown on secondary loop emissions, the variation of parameters 7 (blowdown rate), 9 (decofactor of the blowdown filters) and 5 (steam emissions from the low-pressure section) is initially decisive; these form the first parameter set. The remaining parameters are assumed to be constant for the calculations of block 1. An exception is the decay constant λ_n. Since comparatively long transport times occur particularly in the area of the blowdown demineralization system (compare Section 2.6, Mass Contents, Mass Flows, Steam Emissions), three values of λ_n were assumed for the sensitivity analysis.

With $\lambda_n = 10^{-12}$, 10^{-6}, 10^{0} decays per second, the nuclide spectrum was adequately considered with regard to half-lives. The nuclides Sr 98 with $\lambda = 1.15$ sec^{-1}, I 131 with $\lambda = 9.93 \cdot 10^{-7}$ sec^{-1} and Pn 239 with $\lambda =$ $= 8.97 \cdot 10^{-13}$ sec^{-1} approximately correspond to the selected decay constants.

In comparing the calculations with different decay constants, and in particular in evaluating the indicated overall emission rates, it should be noted that, unless otherwise indicated, the emission calculations were carried out with the primary loop concentration standardized to 1 Ci/t. This standardization serves to improve comparability and makes it possible to calculate

emissions of special nuclides with the same decay constants by simply multiplying [them] by the actual primary loop concentration values. The standardization to a primary loop concentration also means, however, that for a nuclide with $\lambda = 10^{-12}$ sec^{-1} exactly 10^{12} times more nuclei of this nuclide are present than is the case with a nuclide with $\lambda = 1$ sec^{-1}.

To improve comparability, the emission calculations were carried out with a steady-state activity distribution, with the exception of block 6.

4.2. Quantitative Analysis

4.2.1. Block 1 - Blowdown Demineralization

In this study, the effect of the emission parameters which are associated with blowdown demineralization was determined; thus the sensitivity of the emissions from the secondary loop to the parameters

7). ABS - Blowdown rate

9). DF_{ABS} - Decofactor in the blowdown filters

5). Steam emissions from the low-pressure section (steam water degasification system)

was calculated, and in each case this was done for three different decay constants. Table 4.1.1 shows the parameter values or value ranges used, arranged by parameter sets.

The results of the emission calculations for block 1 are found in Tables 4.1.2--4.1.8. Correspondingly, Figures 4.1.2-4.1.8 show the most important calculations in graphic form.

The tables and the above-mentioned figures provide the following insights:

4.2.1.1. Examination of ABS

a. Qualitative Analysis

1. The larger the ABS [blowdown rate], the smaller the emission rate E for $DF_{ABS} \geq 10$.

2. For $DF_{ABS} = 1$, i.e., for the case where the blowdown filters have no retention capability, as ABS increases E rises, and does so more intensely the greater the steam emission D_L from the feedwater degasification system.

3. With otherwise constant conditions, it is not possible to achieve a significant reduction in the emission by increasing the decofactor DF_{ABS} of the blowdown filters beyond a value of 10.

TABLE 4.1.1. PARAMETER DATA FOR BLOCK 1.

Parameter	Value range	Individual value
	Parameter Set 1	
7) ABS in[t/hr]	0 - 24	0,4,8,12,16,20,24
9) DF_{ABS}	1 - 10^4	10^0,10^1,10^2,10^3,10^4
5) D_L in[t/hr]	0 - 6	0,1,2,3,4,5,6
3) λ in[sec^{-1}]	10^{-12} - 10^0	10^{-12},10^{-6},10^0
	Parameter Set 2	
1) A_n in[Ci/t]	-	1,0
2) L in[t/hr]	-	0,02
4) D_H in[t/hr]	-	3,6
6) f_{DE} in [%]	-	0,25
8) f_{ABS} in [%]	-	66
10) α_H	-	10^3
11) α_L	-	10^5

b. Quantitative Analysis

A doubling of the blowdown rate ABS to 16 t/hr causes a 43% reduction in the emission rate E when $\lambda = 10^{-6}$ sec^{-1}.

These data apply essentially to the entire range of values of $DF_{ABS} \geq 10$ and are read off for $DF_{ABS} = 10^3$. The steam emission D_L causes a damping of this reduction. The effect is, however, not very great: a doubling of the steam emtssion D_L from 2 t/hr to 4 t/hr leads to a reduction of 41% instead of the above-given value of 44%.

For the case where $DF_{ABS} = 1$, emission increases by 14% when the ABS is again doubled from 8 to 16 t/hr and $D_L = 2$ t/hr. With twice the steam emission ($D_L = 4$ t/hr), the increase in the emission rate is approximately 33%, i.e., more than twice as large.

c. Interpretation

The fact that the results of the calculations correctly reflect the experimentally demonstrated function of the blowdown demineralization system should be regarded as a verification of the model approach as well as the computer program with respect to block 1 - Blowdown.

The fact that the emission rate is relatively greatly independent of the decofactor DF_{ABS} for $DF_{ABS} > 10$ is only an apparent contradiction of the known absorption mechanisms of the blowdown filters. First we should remember the fact that the decofactor is incorporated hyperbolically in the calculation of the activity distribution (compare Section 2.2), and therefore it is to be expected that above a certain DF_{ABS} value no significant reduction in emission is to be expected. This value lies at approximately 10. In addition, another reason can be deduced from the above-mentioned figures:

As described in Section 9.1, model of the secondary loop of a LWR, the pathway blowdown flash tank → feedwater tank is relevant for the emission calculations; the mass flow corresponding to this pathway (flash tank steam with a 66% water content) is approximately as large as that corresponding to the blowdown itself; this flow serves as heating steam in feedwater degasification (compare Figure 2.1.5).

For the flash tank, the residual moisture alone provides a decofactor of a maximum of 2.

Since the mass flow increases in direct proportion to the blowdown rate ABS via the pathway blowdown flash tank → feedwater tank, as ABS increases the unblowndown, i.e., unfiltered influx to the feedwater tank and ultimately the emission proportion associated with the steam emission D_L thus also increase.

The effect of this bypass pathway is also clearly evident in comparing the emission rate in the event of minimum activity transport via the bypass pathway (residual moisture $f_{ABS} = 0.0$; $\alpha_H = 10^3$) on the one hand and maximum transport ($f_{ABS} = 66.0$; $\alpha_H = 10^3$) via the bypass pathway on the other

(compare Table and Figures 4.1.7 and 4.1.8). Even if there is no feedwater degasification, when this pathway is taken into account the emission rate is doubled, in the case of a maximum degasification rate by a factor of 3.7; both are calculated with full blowdown.

4.2.1.2. Analysis of Steam Emission D_L

a. Qualitative Evaluation

1. Under otherwise constant conditions the emission rate increases as the steam emission rate D_L from feedwater degasification rises.

2. This applies especially during complete blowdown.

b. Quantitative Evaluation

In the case of the maximum rise of steam emission D_L from 0 to 6 t/hr, the increase of the emission rate E in the absence of blowdown (ABS = 0 t/hr) is approximately 80% and in the case of maximum blowdown (ABS = 24 t/hr) approximately 230%.

When steam emission D_L is doubled from 2 t/hr to 4 t/hr, emission at ABS = 0 t/hr increases by approximately 20% and at ABS = 24 t/hr by approximately 43%. These figures apply to $DF_{ABS} = 10^3$ and $\lambda = 10^{-6}$ sec^{-1}.

c. Interpretation

This situation could be expected since any activity in the feedwater tank is emitted to an amplified degree as feedwater degasification increases relative to steam emission D_L.

It is noteworthy that the percentage increase of emission as a result of rising steam emission D_L is always greater with blowdown than without. Thus the increase of emission when D_L is doubled from 2 to 4 t/hr and in the case of complete blowdown (ABS = 24 t/hr) is approximately twice that which occurs without blowdown (ABS = 0 t/hr). This relative increase is due to the blowdown flash tank-feedwater tank pathway (compare analysis of parameter ABS in the previous item).

4.2.1.3. Analysis of Decontamination Factor DF_{ABS}

Discussion of the effects of decofactor DF_{ABS} on the overall emission rate E is found under items 1 and 2.

4.2.1.4. Analysis of Decay Constants

Analysis of the dependency of emission on decay constant λ confirms the importance of the decay correction factor contained in the mathematical algorithm. In this case the nuclide concentrations involved in the emission computation are computed as a function of time, i.e., with regard to radioactive decay. The transport times of the mass flows and dwell times at points are decisive in this case (compare Section 2.6 Mass Contents, Mass Flows and Steam Emissions).

These transport and dwell times are especially long in the area of blowdown demineralization. This leads to a decisive decay correction factor especially for a decay constant of $\lambda = 1\ sec^{-1}$.

4.2.1.5. Summary of Results from Block 1

1. As expected, operating blowdown demineralization (DF ≥ 10) is the most effective parameter with regard to total emission with the parameter stipulation of block 1.

2. Invariance of emission to variations of the decofactor of blowdown demineralization is attributed primarily to the bypass effect of the blowdown flash tank → feedwater tank pathway at values $DF_{ABS} > 10$.

3. Intensified feedwater degasification and associated steam emission D_I in the low pressure section increases the total emission in the area of the parameter constellation of block 1, especially in the case of complete blowdown, by a factor 2.3.

4.2.2. Block 2 - High Pressure Section

In this analysis the sensitivity of emission from the secondary loop to the parameter specified in the high pressure section is determined. They are:

6). Residual moisture f_{SG} in the steam generator.

10). Phase distribution α_H in the high pressure section.

4). Steam emission rate D_H from the high pressure section.

Again the total emission E was selected as the dependent variable. Residual moisture f_{SG} in the steam generator was selected as the independent variable. Here α_H, D_H and λ were used as the variation parameters. Table 4.2.1 shows the values used or value ranges of the parameters arranged by parameter sets.

The following analyses apply primarily to Tables 4.2.2-4.2.9 and the corresponding Figures 4.2.2-4.2.9.

TABLE 4.2.1. PARAMETER DATA FOR BLOCK 2.

Parameter	Value range	Individual values
Parameter set 1		
6) f_{SG} in [%]	0,01 -6,1	$1,0 \cdot 10^{-2}$, $2,5 \cdot 10^{-2}$, $6,25 \cdot 10^{-2}$, $1,56 \cdot 10^{-1}$, $3,91 \cdot 10^{-1}$, $9,77 \cdot 10^{-1}$, 2,44, 6,10
10) α_H	10^{-4} - 10^{+5}	10^{-4}, 10^{0}, 10^{1}, 10^{2}, 10^{3}, 10^{4}, 10^{5}
4) D_H in [t/h]	2 - 10	2, 4, 6, 8, 10
3) λ in [sec^{-1}]	10^{-12} - 10^{-0}	10^{-12}, 10^{-6}, 10^{0}
Parameter set 2		
1) A_n in [Ci/t]	-	1,0
2) L in [t/h]	-	0,02
5) D_L in [t/h]	-	2,0
7) ABS per DE in[t/h]	-	8,0
8) f_{ABS} in [%]	-	66
9) DF_{ABS}	-	10^{4}
11) α_L	-	10^{2}

4.2.2.1. Analysis of Residual Moisture f_{DE}

a. Qualitative Evaluation (compare Table 4.2.2 and Figure 4.2.2).

As residual moisture f_{DE} increases, the emission E also rises. This applies to all $\alpha_H > 10$, but to varying degrees: as expected, gradient $\frac{\partial E}{\partial f}_{FD}$ of emission for $\alpha_H = 10^5$ is the maximum. It decreases for smaller α_H and disappears for $\alpha_H = 1$. Since $\alpha_H = 1$ indicates a resistance-free phase transfer, as it applies to tritium in the chemical form HTO, this correctly reproduced special case is good confirmation of the computer model.

b. Quantitative Evaluation

The emission rate at $\alpha_H = 10^5$ (salt-like nuclide behavior) rises by 5 times the initial value (from $2.52 \cdot 10^{-9}$ to $1.25 \cdot 10^{-8}$ Ci/sec) over the entire range of residual moisture f_{SG} from 10^{-2} to 6.1%. The corresponding increase of emission rate is approximately 1.5 for a nuclide with a phase transition behavior similar to that of iodine ($\alpha_H \sim 10^2$). At no phase transition resistance ($\alpha_H = 1$) the emission rate remains invariant with respect to residual moisture at $9.02 \cdot 10^{-7}$ Ci/sec, as expected.

These data apply to $\lambda = 10^{-6}$ sec^{-1} and $D_H = 4$ t/hr.

c. Interpretation

These computations confirm the known fact that as the water content of the main steam increases, i.e., that of the residual moisture f_{SG}, a decrease of emission rate takes place concurrently and especially more dramatically the more the nuclide under consideration demonstrates a salt-like behavior during the phase transition (α_H increases).

4.2.2.2. Analysis of Steam Emission Rate D_H in the High Pressure Section

a. Qualitative Evaluation

An increase of steam emission D_H leads to increased overall emissions. This applies to all values of residual moisture f_{SG} and the phase distribution factor α_H and decay constant λ.

b. Quantitative Evaluation

Doubling of steam emission rate D_H leads to an emission rate E which is higher by the doubling factor 1.5-1.7 in a lesser dependency (< 10%) on α_H.

When the steam emission rate D_H rises over the entire range of 2-10 t/hr (factor of 5) emission E increases by a factor of 3.0-3.5. The effect of residual moisture f_{SG} hereon is < 1% at small α_H ($\alpha_H = 10$) and < 10% at high α_H ($\alpha_H = 10^5$). The aforementioned values apply to $\lambda = 10^{-6}$ sec^{-1}. In computations with $\lambda = 10^{0}$ sec^{-1} it is noteworthy that here the doubling factor is exactly two.

This is due primarily to the negligibly small contribution of steam emission D_L from the low pressure section to overall emission. The short-lived nuclide (λ = 1 sec^{-1}) does not reach the feedwater tank. It is either emitted with the high pressure emission through the stack or decays enroute to the feedwater tank.

Emissions of a nuclide with noble gas properties ($\alpha_H = 10^{-4}$, all decofactors = = 1) for $\lambda = 10^{-6}$ and 1 sec^{-1} were examined as a special case. The doubling factors for the steam emissions are approximately 1.8 and 2.0. Noble gas percentages which are released through the stack via the exhaust system of the evaporator overhead condenser are higher by a factor of 1000 than the power house emissions.

c. Interpretation

Reduction of steam emission D_H from the high pressure section by one-half reduces the total emission only 1.6-fold. This is due to the fact that the steam emission D_L from feedwater degasification contributes to overall emission, in the aforementioned computations with D_L = 2 t/hr.

4.2.2.3. Analysis of Phase Distribution Factor α_H in the High Pressure Section

a. Qualitative Evaluation

Under otherwise constant conditions the total emissions E decrease as α_H increases. This applies to all residual moistures f_{SG} and steam emission rates D_L.

b. Quantitative Evaluation

An increase of α_H by a factor of 10 is followed by a drop in emission by a factor of 1.1 at high residual moisture f_{SG} = 6.1% and by a factor of 5.4 for the minimum residual moisture f_{SG} = 0.01% and D_H = 4 t/hr, respectively.

An emission decrease by a factor of 2.75 occurs for high residual moisture of f_{DE} = 6.1% over the range $\alpha_H = 10^0$-10^3. On the other hand, it drops by a factor 44.6 at residual moisture f_{DE} = 0.01% which is 600 times less.

c. Interpretation

As was derived and explained in mathematical model of phase distribution and decontamination factors in Section 2.2, the phase distribution factor α_H is a direct measure of the ratio of the specific activities of a certain nuclide in the liquid to the gaseous phase of the mass flow under consideration. Here α_H is an indirect measure for phase transition resistance and as a result is associated with residual moisture transport. Thus $\alpha_H = 10^{-4}$ indicates pure noble gas behavior, α_H = 1 phase-invariant behavior as tritium shows and $\alpha_H \doteq 10^{+4}$ a salt-like behavior, i.e., a high α_H is associated with a high liquid/gas phase transition resistance and dramatic coupling to residual moisture f_{DE}. The considerably higher emission drop by an increased α_H at low residual moisture is thus explained.

4.2.2.4. Analysis of Decay Constants

The effects of decay constants on emission rate E mentioned essentially in unit 1 also occur in the high pressure range. Thus the decay correction factor incorporated in the computer program causes a concentration decrease dependent on transit time, with the effect of diminished emission rates for shorter-lived nuclides.

4.2.2.5. Summary of Results of Block 2

In the three relevant variation parameters, residual moisture f_{DE}, steam emission rate D_H in the high pressure section and phase distribution factor α_H in the high pressure section, the sequence of importance of parameters can no longer be assigned as easily to total emission as in block 1.

-- If the steam emission rate D_H increases from 2 to 10 t/hr, emission higher by a factor of 3-3.5 must be expected.

-- If the residual moisture f_{DE} increases from 0.15% to 6.1%, in the case of nuclides with high α_H, especially therefore in ionically present nuclides, an emission increase by a factor of 10.7 must be expected.

-- The total emission E is related to low residual moisture in an especially sensitive manner with regard to high α_H values, i.e., ionically occurring nuclides. At low α_H values the absolute values of emission rates are higher from the start (approximately $5 \cdot 10^{-7}$ Ci/sec). The relative difference of emission, however, is comparably small with regard to α_H values of differing magnitude.

-- Noble gases are discharged for the most part (approximately 99%) via the exhaust system of the evaporator overhead condenser through the stack.

Upon comparison of computations with different decay constants, especially in the assessment of the stipulated total emission rates, it must be borne in mind that emission computations, unless otherwise specified, were carried out with primary loop concentration normalized to 1 Ci/t. The normalization is used for better comparability and allows computation of emissions of special nuclides of uniform decay constants by simple multiplication with the actual primary loop concentration values.

Normalization to a primary loop concentration, however, means that exactly 10^{12} times more nuclei of a nuclide with $\lambda = 10^{-12}$ sec^{-1} occur than in the case of a nuclide with $\lambda = 1$ sec^{-1}.

4.2.3. Block 3 - Low Pressure Section

The parameters specified for the low pressure section, the effect of which on emission is determined above the power house roof, are:

10). Phase distribution factor α_H in the high pressure section.

5). Steam emissions D_L from the low pressure section (feedwater degasification).

7). Blowdown rate ABS.

11). Phase distribution factor α_L in the low pressure section or ratio α_L/α_H.

The dependent variable, total emission rate E, was determined as a function of the independent variables α_H and DL. Here the ratio α_L/α_H and blowdown rate ABS were used as parameters.

The phase distribution factor α_H was included in block 3 since it plays an important role for the water degasified in the feedwater tank.

Table 4.3.1 shows the values used or the ranges of parameters arranged by parameter sets.

A selection of computations is shown in the following. The cited analyses refer to Tables 4.3.2-4.3.4 and Figures 4.3.2-4.3.4, which correspond to them.

4.2.3.1. Analysis of Steam Emission Rates D_L or Feedwater Degasification
(compare Table 4.3.2 and Figure 4.3.2)

a. Qualitative Evaluation

As steam emission rate D_L increases, the total emissions rise as a function of α_H.

b. Quantitative Evaluation

Emission E increases over the entire range of variation of D_L of 0-6 t/hr by a factor 1.7 at α_H = 1 to 2.8 at $\alpha_H = 10^5$. The doubling factors (= emission increase as a result of doubling of the parameter) are between 1.1 and 1.5. These data apply to α_L/α_H = 100 and an average blowdown rate ABS of 12 t/hr.

c. Interpretation

The fact that 100% increase of steam emission rate D_L from the low pressure section results in only a (α_H-dependent) increase of emission rate of 10-50% is due to the predominant high pressure emission (compare Section 4.2.2 block 2). The more dramatic effect of an increase of D_L at high α_H derives on the hand from the higher concentration in the steam generator water, i.e., a higher activity flow over the course of blowdown, and on the other hand from the higher percentage of residual moisture content which decisively affects the activity influx from the high pressure section to the feedwater tank.

4.2.3.2. Analysis of High Pressure Phase Distribution Factor α_H
(compare Table and Figure 4.3.3 and 4.3.4)

a. Qualitative Evaluation

As α_H increases the total emission E decreases, most radically in the area of α_H = 100 and moreso for high blowdown rates than for low ones.

TABLE 4.3.1. PARAMETER DATA FOR BLOCK 3.

Parameter	Range	Individual values
Parameter set 1, variable		
10) α_H	10^0 - 10^5	$10^0, 10^1, 10^2, 10^3, 10^4, 10^5$
11) α_L/α_H	10^0 - 10^3	$10^0, 10^1, 10^2, 10^3$
5) D_L in [t/h]	0 - 6	0, 1, 2, 3, 4, 5, 6
7) ABS in [t/h]	0 - 24	0, 6, 12, 18, 24
Parameter set 2, fixed		
1) A_n in [Ci/t]	-	1,0
2) L in [t/h]	-	0,02
3) λ in [sec^{-1}]	-	10^{-5}
4) D_H in [t/h]	-	3,6
6) f_{SG} in [%]	-	0,25
8) f_{ABS} in [%]	-	66
9) DF_{ABS}	-	10^4

b. Quantitative Evaluation

The total emission decreases by a factor of 17.3 without blowdown and by a factor of 182 at maximum ABS = 24 t/hr per steam generator over the entire range of α_H (5 orders of magnitude).

In the range of α_H = 100 a change of α_H by a factor of 10 is associated with a reduction factor of emission of 3.2 at ABS = 0 t/hr and 7.2 at ABS = = 24 t/hr. On the other hand, at α_H = 1 or α_H = 10^4 the aforementioned factors are 1.3 and 1.0 for ABS = t/hr.

c. Interpretation

It was also found here that a reduction of emission of nuclides which occur ionically under reactor conditions (α_H large) by means of blowdown is more effective than in the case of nuclides with smaller α_H (compare discussion of the parameter, steam emission rate D_L, in previous Section 4.2.3.1).

The saturation behavior of the emission at high α_H values derives from the activity transport which then almost exclusively determined by residual moisture. Saturation behavior at small α_H and in the absence of blowdown is caused by high steam transport percentage.

4.2.3.3. Analysis of Blowdown Rate ABS

The recognized effect of blowdown on emission rates is apparently consistent with previous determinations in block 3 as well. The reduction factor of the emission rates is 4.1 for α_H = 1 and 43.3 for α_H = 10^5 over the entire range of blowdown rates of ABS = 0-24 t/hr.

Blowdown is approximately ten times more effective in the case of nuclides which occur ionically under reactor conditions than in the case of nuclides with α_H = 1 (for example tritium). The doubling factor is between 1.4 and 1.7 depending on α_H. The values are stipulated for D_L = 2 t/hr.

4.2.3.4. Analysis of the α_L/α_H Ratio

According to the authors Jonas, Class and Styrikovich (compare bibliography of Section 2.5 Decontamination Factors) the phase distribution factor α is a function of the state of the steam or liquid phase.

It is assumed:

$$\alpha \sim \left(\frac{\rho'}{\rho''}\right)^{\varkappa}$$

with ρ' = Density of liquid flow,

ρ'' = Density of steam phase,

$\varkappa$ = Exponential factor material-specific.

If now we analyze the behavior of the phase distribution factors α_L/α_H and assume by way of simplification that $\mathcal{K}_L = \mathcal{K}_H$ applies, we find $10 < \alpha_L/\alpha_H < 10^3$ for the operating states under analysis here.

Since we know of no relevant measurements or data, the α_L/α_H ratio was incorporated and computed as a variation parameter. It is immediately apparent from Figure 4.3.4 that the effect on emission rate is not noticeable. Therefore in all other computations it was assumed $\alpha_L/\alpha_H = 100$.

4.2.3.5. Summary of Results of Block 3

At the blowdown rate ABS the doubling factors are somewhat higher than for the steam emission rate. Phase distribution factor α_H in the high pressure section is especially important to emission rates in an interaction with blowdown. The phase distribution factor α_L in the low pressure section affects emission computations to a very minor degree since the decontamination factors in the low pressure section are determined largely by the high residual moistures.

4.2.4. Block 4 - Standard Computation

The variation parameters in this block 4 are the decay constant λ, and primary loop concentration A.

Thus a quick survey of the expected secondary loop emissions (above the roof) in the case of special nuclides $n(\lambda_n)$ and primary loop activities A_n is possible.

On the other hand, the computer program is being checked since emission rates to be computed must be proportional to the primary loop activities.

Table 4.4.1 shows the values used and the ranges of the parameters arranged by parameter sets.

Table 4.4.2 shows emission computations for primarily ionically occurring nuclides ($\alpha_H = 10^3$), Table 4.4.3 shows the emission rates for gases ($\alpha = 10^{-4}$).

TABLE 4.4.1. PARAMETER DATA FOR BLOCK 4.

Parameter	Range	Individual values
	Parameter set 1	
1) A_n in [Ci/t]	10^{-8} -10^{1}	10^{-8},10^{-7},10^{-6},10^{-5},10^{-4}, 10^{-3},10^{-2},10^{-1},10^{0},10^{1}
3) λ in [sec^{-1}]	10^{-12}- 10^{0}	10^{-12},10^{-10},10^{-8},10^{-6} 10^{-4},10^{-3},10^{-2},10^{0}
10) α_H -	10^{-4} - 10^{3}	10^{-4},10^{3}
	Parameter set 2	
1) L in [t/h]	-	0,02
4) D_H in [t/h]	-	3,6
5) D_L in [t/h]	-	2,0
6) f_{SG} in [%]	-	0,25
7) ABS per DE in [t/h]	-	8,0
8) f_{ABS} in [%]	-	66
9) DF_{ABS}	-	10^{4}
10) α_H	-	10^{3}
11) α_L	-	10^{2}

4.2.4.1. Analysis of Primary Loop Concentration A

As was expected, the primary loop concentration A is involved linearly in the emission computations, i.e., the emission rate E is directly proportional to the primary loop activity.

4.2.4.2. Analysis of Decay Constant λ

a. Qualitative Evaluation

The greater λ, i.e., the more short-lived the nuclide under consideration, the smaller the emission rates become if $10^{-6} \leq \lambda \leq 1$.

b. Quantitative Evaluation

The emission rate above the power house roof decreases by a factor $8.1 \cdot 10^6$ in the case of salts and $2.4 \cdot 10^4$ in the case of gases over the range for λ of 10^{-6} - 1 which extends over six orders of magnitude (in the case of secondary loop-induced stack emissions, the factor is approximately $8.1 \cdot 10^4$).

The increment from $\lambda = 10^{-2}$ sec^{-1} to $\lambda = 1$ sec^{-1} is especially salient for the secondary loop. This rise factor 100 for λ causes a decrease of emission rates by a factor $1.3 \cdot 10^4$ for salts and $6.5 \cdot 10^3$ for gases (power house emissions) and $2.4 \cdot 10^4$ (secondary loop stack emission). The tenth increment from $\lambda = 10^{-3}$ sec^{-1} to $\lambda = 10^{-2}$ sec^{-1} decreases the power house emission rate by a factor of 12 for salts and 2.5 for gases.

c. Interpretation

The recession of emission is especially dramatic in the range from $\lambda = 1$ sec^{-1} to $\lambda = 10^{-2}$ sec^{-1}. This situation results, as explained previously, from the decay correction factor for the stipulated plant-specific transport times.

4.2.4.3. Analysis of Phase Distribution Factor α_H

The difference between the largely ionically and gaseous occurring nuclides mentioned in the previous item with regard to computed emission rate derives from the completely different behavior of salts and gases in the steam generator:

Salts remain for the most part in the aqueous phase; gases, on the other hand, in the vapor phase. This means considerably higher dwell times in the steam generator for salt than for decontamination-free gases.

4.2.4.4. Summary of Results of Block 4

The computer program yields the linear relationship between emission rate and primary loop activity in a correct manner. It also shows that the so-called standard parameter values constitute a suitable basis for parameter set 2 used in the individual units.

Particularly the computer program yields the expected varied behavior of gases and salt nuclides.

Finally, the computer program clearly shows the relationship between plant-induced transport times and decay correction factor.

4.2.5. Block 5 - Extreme Computation

The computations reproduced in the following in excerpt form are used to check the program even in extreme parameter ranges which exceed the standard ranges (compare previous blocks 1 through 4). The following were varied as parameters important for this purpose:

2). Leakage rate L in [t/hr].

4). High pressure steam emission D_H in [t/hr].

6). Residual moisture f_{SG} in steam generator in [%].

10). Phase distribution factor α_H.

Table 4.5.1 shows the values used or the ranges of the parameters arranged by parameter sets.

The independent variables, leakage rate L, residual moisture f_{SG} and phase distribution factor α_H as well as the variation parameters α_H, f_{SG} and D_H are compared to the dependent variable, emission rate E.

4.2.5.1. Analysis of Leakage Rate L

As is apparent from the figures and Tables 4.5.2-4.5.6, the expected directly proportional dependency of emission rate E on leak rate L is accurately reproduced by the computer program.

4.2.5.2. Analysis of Residual Moisture in Steam Generator f_{SG}

The residual moisture behavior of emission rates discussed in previous blocks is also shown in the present extreme computations. In this case two saturation ranges become clear:

1. For $\alpha_H = 10$ a change of emission rate as residual moisture values decrease from $f_{SG} \leq 10\%$ is still rather low.

2. For $\alpha_H > 100$ emission rates for high values of f_{SG} proceed into saturation.

This saturation behavior can be easily explained by the activity transport determined predominantly by residual moisture for $\alpha_H > 100$ and activity transport associated primarily with steam for $\alpha_H < 10$.

At an increase of residual moisture f_{SG} from the design value 0.25% to 31.2%, an increase of emission rate E by a factor of approximately 4.3 to a value of $9.5 \cdot 10^{-8}$ Ci/sec can be expected for salt-like nuclides. For nuclides with $\alpha_H = 10$ the increase of emission rate E is only approximately 40% at the considerably higher absolute value of $E = 8.74 \cdot 10^{-7}$ Ci/sec.

TABLE 4.3.1. PARAMETER DATA FOR BLOCK 5.

Parameter	Range	Individual values
Parameter set 1, variable		
2) L in [t/h]	10^{-1} - 10	0,1, 1, 10
4) D_H in[t/h]	1 - 10^2	1, 10, 10^2
6) f_{SG} in[%]	0,25- 31,2	0,25, 1,25, 6,25, 31,2
10) α_H	1 - 10^5	$1,10^1,10^2,10^3,10^4,10^5$
Parameter set 2, fixed		
1) A_n in [Ci/t]	-	1,0
3) λ in [sec^{-1}]	-	10^{-6}
5) D_L in [t/h]	-	2,0
7) ABS in [t/h]	-	24,0
8) f_{ABS} in [%]	-	66,0
9) DF_{ABS}	-	10^4
11) α_L	-	10^2

4.2.5.3. Analysis of Phase Distribution Factor α_H in High Pressure Section

The effect of parameter α_H on emission rate E has already been explained qualitatively under 4.2.5.2. At a constant high pressure steam emission rate D_H = 10 t/hr, E increases by a factor of approximately 32 because of a decrease of the α_H value from 10^5 to 10 at a design residual moisture of f_{SG} = 0.25% and by a factor of approximately 9 at f_{SG} = 31.2% (compare Figures 4.5.4-4.5.6). This situation can also be clarified by comparing the activity transport in the steam and in the residual moisture.

4.2.5.4. Analysis of the High Pressure Steam Emission Rate D_H
(Compare Figure and Table 4.5.7-4.5.9)

The increase of emission rate E due to increase of steam emission rate D_H by a factor of 10 from 10 t/hr to 100 t/hr constitutes a factor of approximately 7-8 depending on residual moisture and α_H values. At this point, among specifically computed examples, let us cite computation of tritium emissions as a function of residual moisture f_{SG} and steam emission in the high pressure section (compare Figure and Table 4.5.10). The primary loop activity A of tritium was assumed to be A = 0.5 Ci/t and the decay constant $\lambda = 1.78 \cdot 10^{-9}$ sec^{-1}.

The residual moisture-invariant tritium emission rate rises by a factor of 1.8 to $E = 1.01 \cdot 10^{-4}$ Ci/sec when D_H increases from 1 t/hr to 10 t/hr, and by a factor 1.3 to $E = 1.33 \cdot 10^{-4}$ Ci/sec when D_H increases from 10 t/hr to 100 t/hr.

4.2.5.5. Summary of Results of Block 5

An increase of leak rate dramatically affects emission rate, as expected. An increase of steam emission by a factor of 10 causes an emission rate rise by a factor of 7-8. An increase of residual moisture from the design value to 31.2% entails an increase of emission rate by a factor of 4.3.

Two nuclides which differ by three orders of magnitude in the α_H value (α_H = 10 and $\alpha_H = 10^4$) are emitted with varying degrees of intensity with a factor of 9-32, depending on residual moisture.

4.2.6. Block 6 - Spike Computation and Load Change

In this block the computer program is checked for its capability to compute time-dependent phenomena such as an activity spike and load changes.

Since spikes cause, among other things, load changes or pressure transients, spike computations were coupled to the load changes.

A slow load change (10 or 20 hrs) and thus slow mass flow changes were selected since the effect of spike factors on nuclide specific emissions was of primary interest.

Variation computations with specific data sets of nuclides I 131, Co 60 and Co 58 due to their radio-ecologic importance are carried out in block 6.

The emission rates of these nuclides are thus computed for the average and maximum spike cited in Table 2.3.1.2 at a constant spike duration for the load change from 100% to 85% power in 10 hrs and 20 hrs.

Table 4.6.1 shows the values used and the ranges of parameters arranged by parameter sets.

Figures 4.6.2-4.6.7 show the spike computations.

4.2.6.1. Comparison of Results for Different Time Variation of Load Change

A comparison of load drop from 100% to 85% power in 10 hrs or 20 hrs shows no significant difference in emission rates for any of the three nuclides.

A difference would only be expected if varied spike height had been assumed based on different load changes.

4.2.6.2. Comparison of Spike Heights

By definition the nuclide concentration in the primary loop is increased by the spike factor under spike conditions. It can therefore be expected that comparison of emission rates of nuclides for mean or maximum spike reproduces the behavior of these two spiking factors.

In fact, the time-dependent computations accurately reproduce this relationship. Thus the maximum emission rate for iodine 131 at a maximum spiking factor of 100 is greater exactly by a factor of 1.92 than for a mean spiking factor of 52. The same applies to nuclides Co 60 and Co 58.

The decay period of the spike-induced emission rates according to which the two spike computations assume the same value is also of interest. This period for iodine 131 is 20 hrs, for Co 60 - 35 hrs and for Co 58 - 45 hrs. These time data must be analyzed in relation to the assumed spike duration which was arbitrarily fixed at 2 hrs for iodine 131 and 10 hrs for the cobalt isotopes in order to test the program. The actual spike durations are shown in Table 2.3.1.2.1.

4.2.6.3. Comparison of Nuclides with Regard to Spike

A peaked or plateau-like variation occurs at the emission rate depending on the spike duration.

All three nuclides show the expected exponential decay of emission rates from the end of spike.

In the case of a short spike duration, the maximum of the cumulative emissions is achieved earlier, as expected, and its decrease proceeds more rapidly.

The maximum emissions achieved with regard to radiologic decay in the atmosphere (computation with mean spike) are notable and reflect the complex relationships:

$$\text{I } 131\text{: } 2.73 \cdot 10^{-2} \text{ Ci}$$

$$\text{Co } 60\text{: } 7.47 \cdot 10^{-4} \text{ Ci}$$

$$\text{Co } 58\text{: } 3.41 \cdot 10^{-2} \text{ Ci}$$

TABLE 4.6.1. PARAMETER DATA FOR BLOCK 6.

Parameter	Range	Individual values	
Parameter set 1, variable			
	Iodine 131	Cobalt 60	Cobalt 58
1) A_n in [Ci/t]	1,0	$2,0 \cdot 10^{-3}$	$4,4 \cdot 10^{-3}$
3) λ in [sec^{-1}]	$9,98 \cdot 10^{-7}$	$4,17 \cdot 10^{-9}$	$1,13 \cdot 10^{-7}$
Spike height mean/max.	52/100	100/500	$2,2 \cdot 10^{3}/10^{4}$
Spike duration in hrs.	2	10	10
10) α_H	10^3	10^3	10^3
11) α_L	10^5	10^5	10^5
Load change	100%/85%	100%/85%	100%/85%
Shutdown time in hrs.	10,20	10,20	10,20
Parameter set 2, fixed			
2) L in [t/h]	-	0,02	
4) D_H in [t/h]	-	3,6	
5) D_L in [t/h]	-	2,0	
6) f_{SG} in [%]	-	0,25	
7) ABS per DE in [t/h]	-	8,0	
8) f_{ABS} in [%]	-	66	
9) DF_{ABS}	-	10^4	

as are the maximum emission rates (likewise medium spikes):

I 131: $3.19 \cdot 10^{-7}$ Ci/sec

Co 60: $2.10 \cdot 10^{-9}$ Ci/sec

Co 58: $1.01 \cdot 10^{-7}$ Ci/sec .

These data should be considered in conjunction with parameter constellation of block 6 (standard parameter values and spike or load change-induced assumptions).

4.2.6.4. Summary of Results of Block 6

The computations of block 6 show that the computer program can compute time-dependent emissions as well in conformity with the system analysis.

4.3. Summary

The foregoing sensitivity analysis of the most important specification factors for emission above the power house roof have shown the following:

1. The computer program developed on the basis of the described systems analysis will accommodate the relative parameters in a satisfactory manner and accurately describe their dynamic interactions, especially with regard to emission, in an adequate manner.

2. Depending on the physical-chemical behavior of the nuclide, the residual moistures in the steam generator and the steam emission from the high pressure portion as well as the blowdown rate are important specification factors for secondary loop emissions.

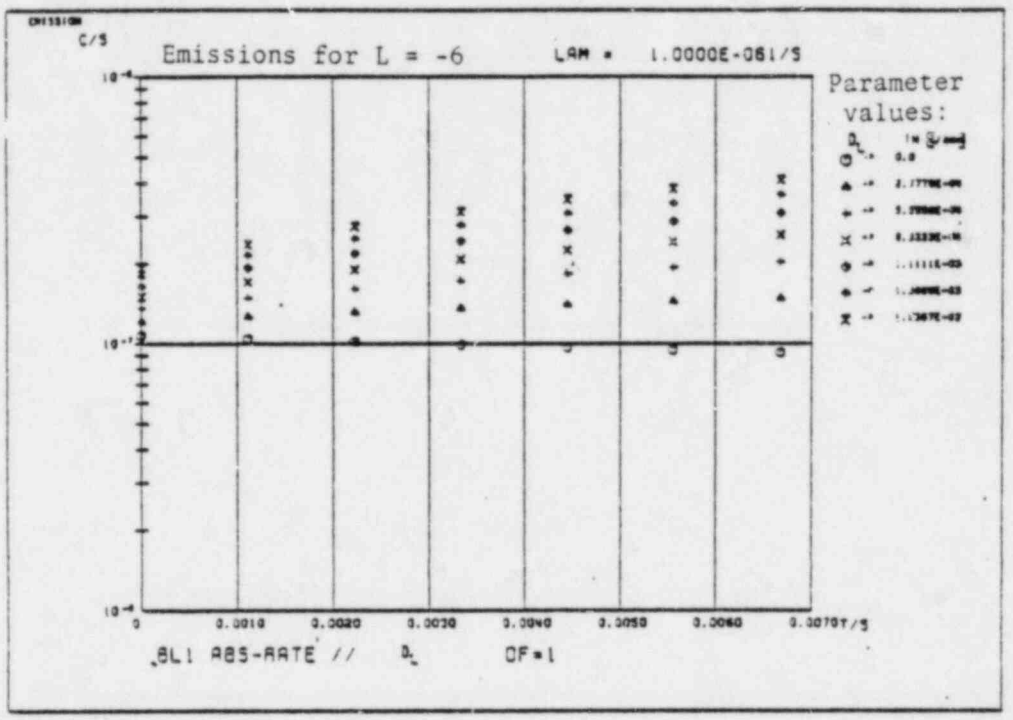
3. In the computation of power house roof emissions, the feedwater degasification contributes a noticeable fraction, especially via the bypass effect of blowdown demineralization flash tank [sic]. Consideration of this effect itself without steam emission from the low pressure section (feedwater degasification) during complete blowdown entails doubling of emission rate and an increase by a factor of 3.7 at maximum feedwater degasification.

4. The emission rate is directly proportional to parameters primary loop activity, spiking factor and leak rate, as expected.

5. It is possible based on these computations to estimate emissions from comparable reactor installations according to corresponding parameter comparison and quantify the estimates by means of the program.

4.4. Tables and Figures

The most important tables and figures for the computations of emission rates under the different parameter constellations of the six parameter blocks follow.

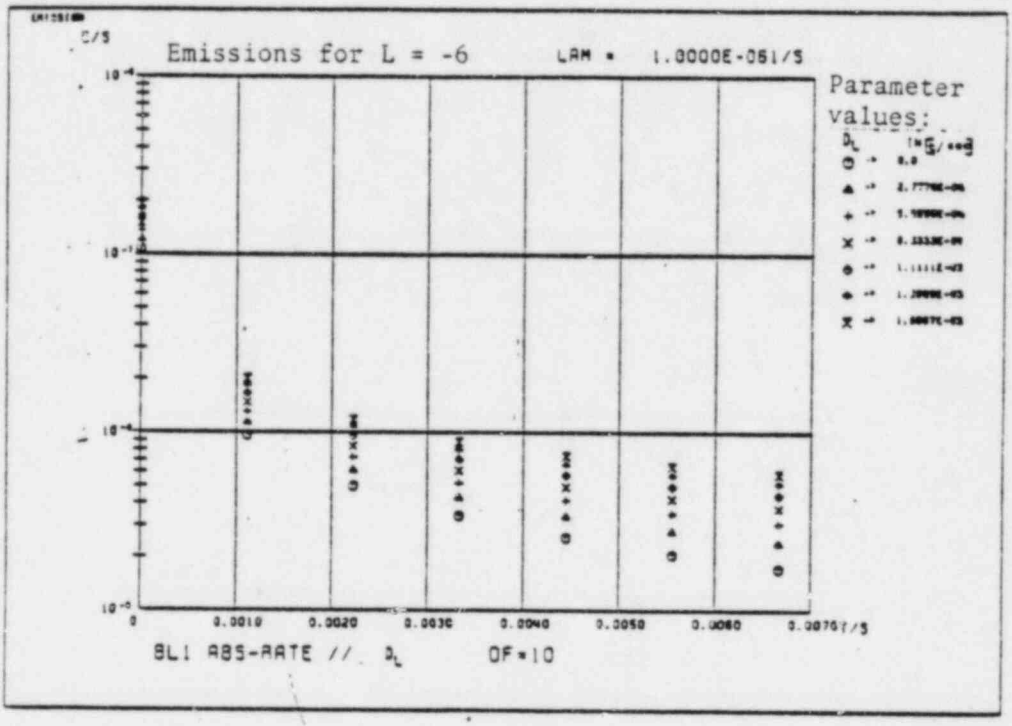


[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:
DFAL = 1.0000E+00 dimensionless
BL1 ABS-RATE

BP (1,10) \ BP (9,1)	0.0	2.78E-04	5.56E-04	8.33E-04	1.11E-03	1.39E-03	1.67E-03
0.0	-1.07E-07	-1.21E-07	-1.36E-07	-1.50E-07	-1.64E-07	-1.78E-07	-1.92E-07
1.11E-03	-1.04E-07	-1.18E-07	-1.46E-07	-1.70E-07	-1.92E-07	-2.13E-07	-2.35E-07
2.22E-03	-1.01E-07	-1.31E-07	-1.60E-07	-1.44E-07	-2.19E-07	-2.46E-07	-2.74E-07
3.33E-03	-9.90E-08	-1.16E-07	-1.72E-07	-2.03E-07	-2.42E-07	-2.77E-07	-3.11E-07
4.44E-03	-9.67E-08	-1.40E-07	-1.82E-07	-2.24E-07	-2.65E-07	-3.06E-07	-3.46E-07
5.56E-03	-9.45E-08	-1.44E-07	-1.92E-07	-2.40E-07	-2.87E-07	-3.33E-07	-3.78E-07
6.67E-03	-9.24E-08	-1.44E-07	-2.01E-07	-2.55E-07	-3.07E-07	-3.56E-07	-4.08E-07

Figure and Table 4.1.2. Block 1 - Blowdown Demineralization. Emission rates as a function of blowdown rate ABS for variable steam emission rates D_L and DF_{ABS} = 1.

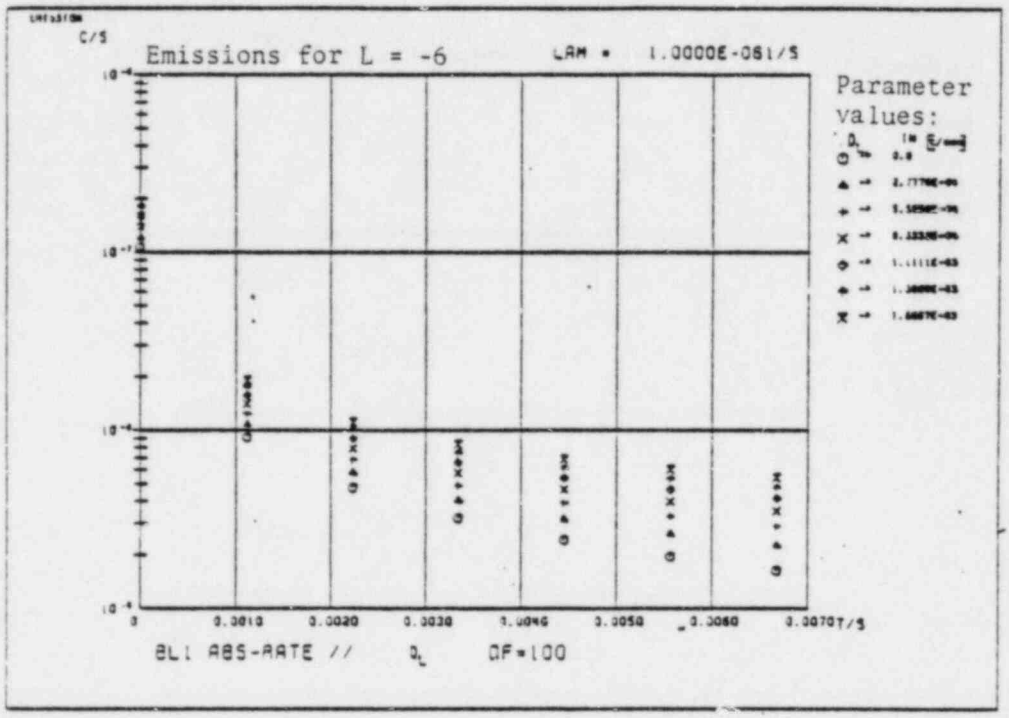


[E- = 10^{-}]

Control parameter
PLOTCONTROL=11 NOKCUN= 2
Emissions in curies per second
For nuclide: L = -6
Parameter list: DFAL = 1.0000E+01 dimensionless
BL1 ABS-RATE

	INP (1,10) IN T/S					
INP (1,10)	0.0	2.78E-04	5.56E-04	8.33E-04	1.11E-03	1.39E-03	1.67E-03
INP (9,13) IN T/S							
0.0	-1.07E-07	-1.21E-07	-1.35E-07	-1.50E-07	-1.64E-07	-1.78E-07	-1.92E-07
1.11E-03	7.55E-09	1.19E-08	1.37E-08	1.58E-08	1.65E-08	1.82E-08	1.99E-08
2.22E-03	5.00E-09	8.11E-09	7.25E-09	9.37E-09	9.49E-09	1.06E-08	1.17E-08
3.33E-03	3.39E-09	4.39E-09	5.20E-09	6.11E-09	7.01E-09	7.92E-09	8.82E-09
4.44E-03	2.56E-09	3.3[illegible]E-09	4.15E-09	4.94E-09	5.74E-09	6.53E-09	7.32E-09
5.56E-03	2.06E-09	2.7[illegible]E-09	3.5[illegible]E-09	4.24E-09	4.9[illegible]E-09	5.69E-09	6.41E-09
6.67E-03	1.72E-09	2.[illegible]E-09	3.[illegible]E-09	3.76E-09	4.[illegible]E-09	5.12E-09	5.80E-09

Figure and Table 4.1.3. Block 1 - Blowdown Demineralization. Emission rates as a function of blowdown rate ABS for variable steam emission rates D_L and DF_{ABS} = = 10.



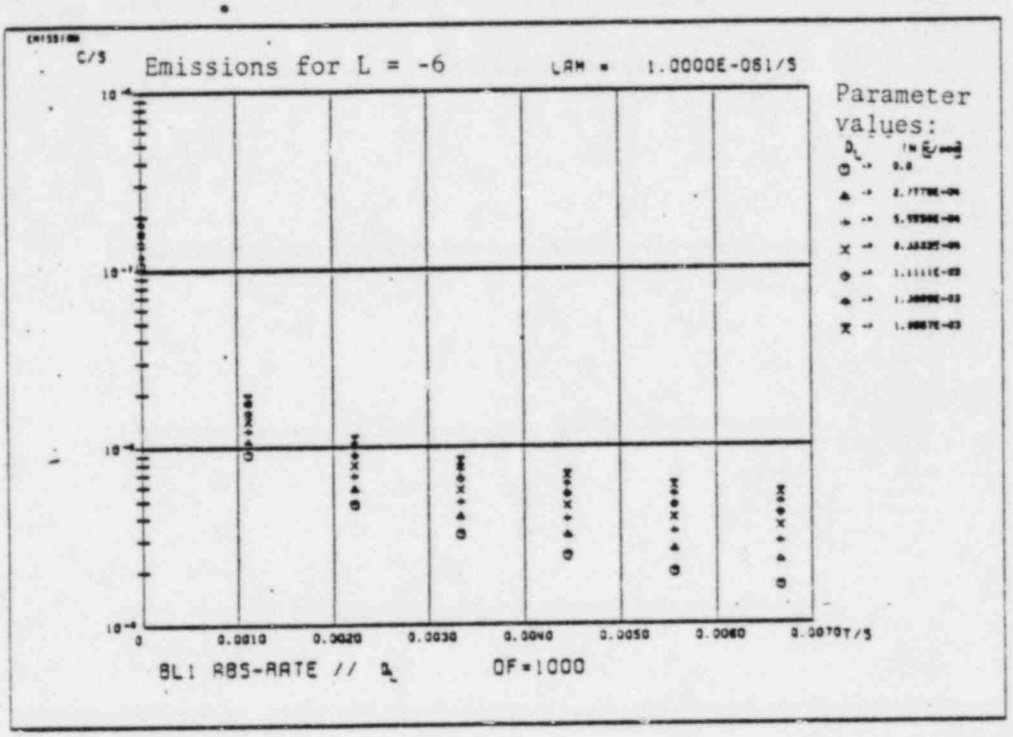
[E- = 10^{-}]

Control parameter
PLOTCONTROL=11 MOKCOL= 2

Emissions in curies per second
For nuclide: L = -6
Parameter list: DFAL = 1.0000E+02 dimensionless
BLI ABS-RATE

	NP (1,10) IN T/S	NP (9,12) IN T/S				
NP (1,10)	0.0	2.78E-04	5.56E-04	8.33E-04	1.11E-03	1.39E-03	1.67E-03
0.0	-1.07E-07	-1.21E-07	-1.35E-07	-1.50E-07	-1.64E-07	-1.78E-07	-1.92E-07
1.11E-03	9.13E-09	1.06E-08	1.24E-08	1.40E-08	1.57E-08	1.73E-08	1.89E-08
2.22E-03	4.77E-09	5.62E-09	6.87E-09	7.92E-09	8.98E-09	1.00E-08	1.11E-08
3.33E-03	3.23E-09	4.07E-09	4.74E-09	5.76E-09	6.80E-09	7.45E-09	8.29E-09
4.44E-03	2.44E-09	3.14E-09	3.91E-09	4.65E-09	5.39E-09	6.12E-09	6.86E-09
5.56E-03	1.96E-09	2.63E-09	3.30E-09	3.96E-09	4.65E-09	5.32E-09	5.99E-09
6.67E-03	1.64E-09	2.27E-09	2.89E-09	3.51E-09	4.15E-09	4.79E-09	5.41E-09

Figure and Table 4.1.4. Block 1 - Blowdown Demineralization Emission Rates as a Function of Blowdown Rate ABS for Variable Steam Emission Rate D_L and DF_{ABS} = 100.



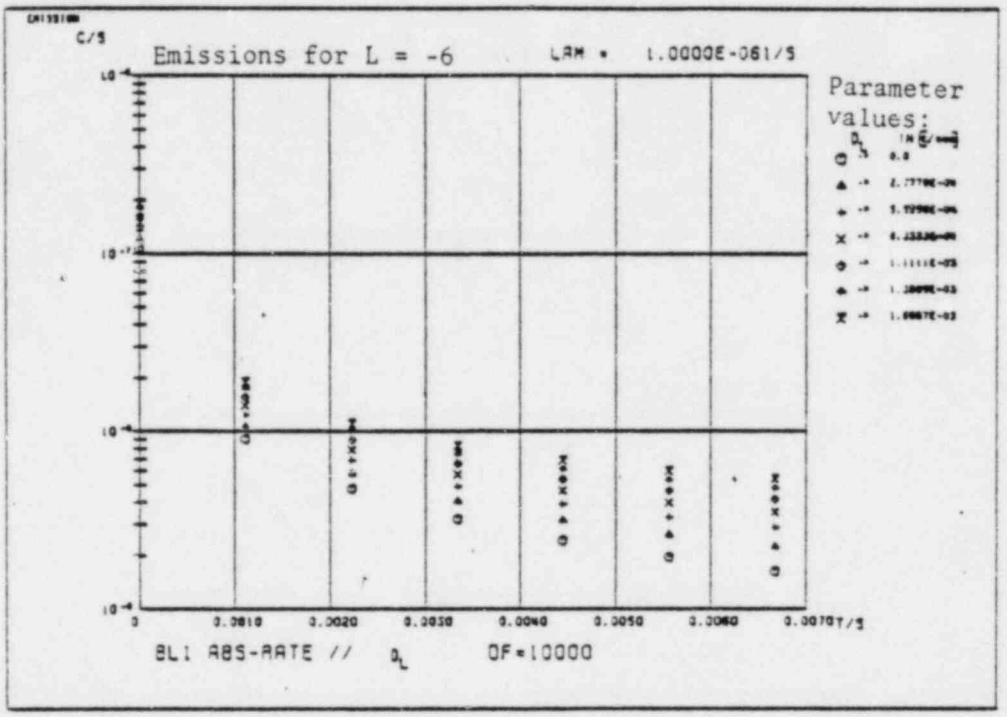
[E- = 10^{-}]

Control parameter
PLOTCONTROL=11 NUMCON= 2

Emissions in curies per second
For nuclide: L = -6
Parameter list: DFAL = 1.00000E+03 dimensionless
BL1 ABS-RATE

BP (1,10)	0.0	2.78E-04	5.56E-04	8.33E-04	1.11E-03	1.39E-03	1.67E-03
0.0	-1.07E-07	-1.21E-07	-1.34E-07	-1.50E-07	-1.64E-07	-1.78E-07	-1.92E-07
1.11E-03	9.09E-09	1.07E-08	1.23E-08	1.40E-08	1.56E-08	1.72E-08	1.89E-08
2.22E-03	4.75E-09	5.79E-09	6.84E-09	7.88E-09	8.93E-09	9.97E-09	1.10E-08
3.33E-03	3.21E-09	4.05E-09	4.89E-09	5.73E-09	6.57E-09	7.40E-09	8.24E-09
4.44E-03	2.43E-09	3.16E-09	3.89E-09	4.62E-09	5.36E-09	6.09E-09	6.82E-09
5.56E-03	1.95E-09	2.62E-09	3.29E-09	3.95E-09	4.62E-09	5.29E-09	5.95E-09
6.67E-03	1.63E-09	[illegible]	[illegible]	[illegible]	[illegible]	4.75E-09	5.37E-09

Figure and Table 4.1.5. Block 1 - Blowdown Demineralization Emission Rates as a Function of Blowdown Rate ABS for Variable Steam Emission Rate D_L and DF_{ABS} = 1000.



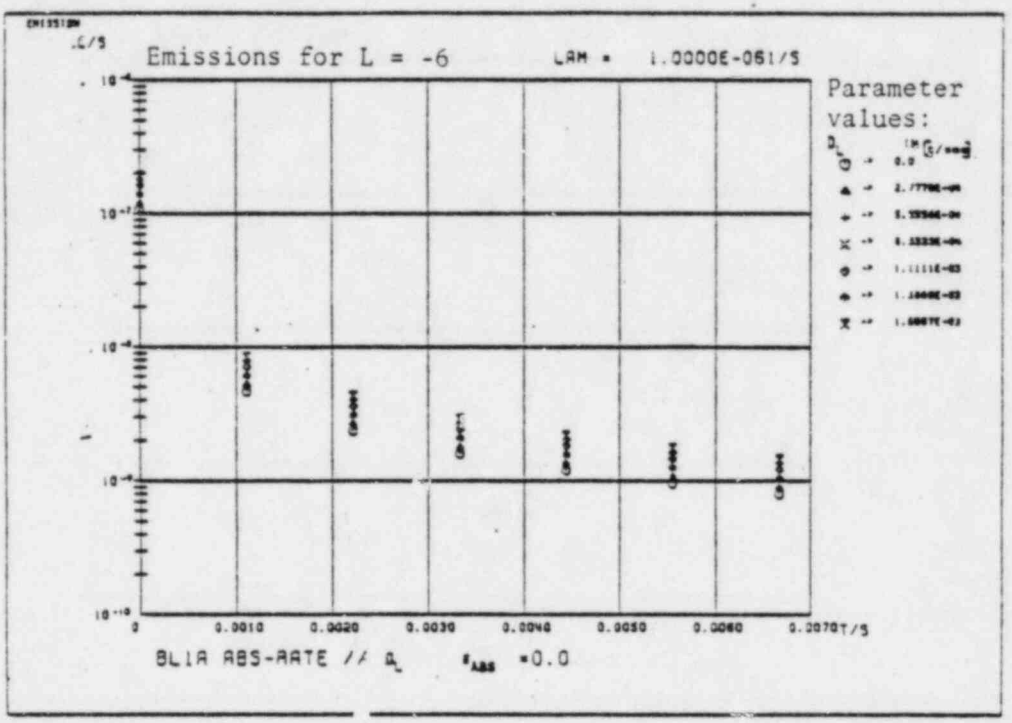
[E- = 10^{-}]

Control parameter
PLOTCONTROL= 11 NOMCOM= 2

Emissions in curies per second
For nuclide: L = -6
Parameter list: CFAC = 1.0000E+04 dimensionless
BL1 ABS-RATE

SP (1,10) / SP (9,13) IN T/S	0.0	2.78E-04	5.56E-04	8.33E-04	1.11E-03	1.39E-03	1.67E-03
SP (1,10) 0.0	-1.37E-07	-1.21E-07	-1.36E-07	-1.50E-07	-1.64E-07	-1.78E-07	-1.92E-07
1.11E-03	9.06E-09	1.07E-08	1.23E-08	1.40E-08	1.56E-08	1.72E-08	1.88E-08
2.22E-03	4.74E-09	5.73E-09	6.84E-09	7.88E-09	8.92E-09	9.97E-09	1.10E-08
3.33E-03	3.21E-09	4.05E-09	4.89E-09	5.73E-09	6.56E-09	7.40E-09	8.24E-09
4.44E-03	2.43E-09	3.14E-09	3.83E-09	4.42E-09	5.35E-09	6.08E-09	6.81E-09
5.56E-03	1.95E-09	2.54E-09	3.29E-09	3.85E-09	4.61E-09	5.28E-09	5.95E-09
6.67E-03	1.63E-09	2.25E-09	2.85E-09	3.50E-09	4.12E-09	4.74E-09	5.37E-09

Figure and Table 4.1.6. Block 1 - Blowdown Demineralization Emission Rates as a Function of Blowdown Rate ABS for Variable Steam Emission Rate D_L and DF_{ABS} = 10,000.



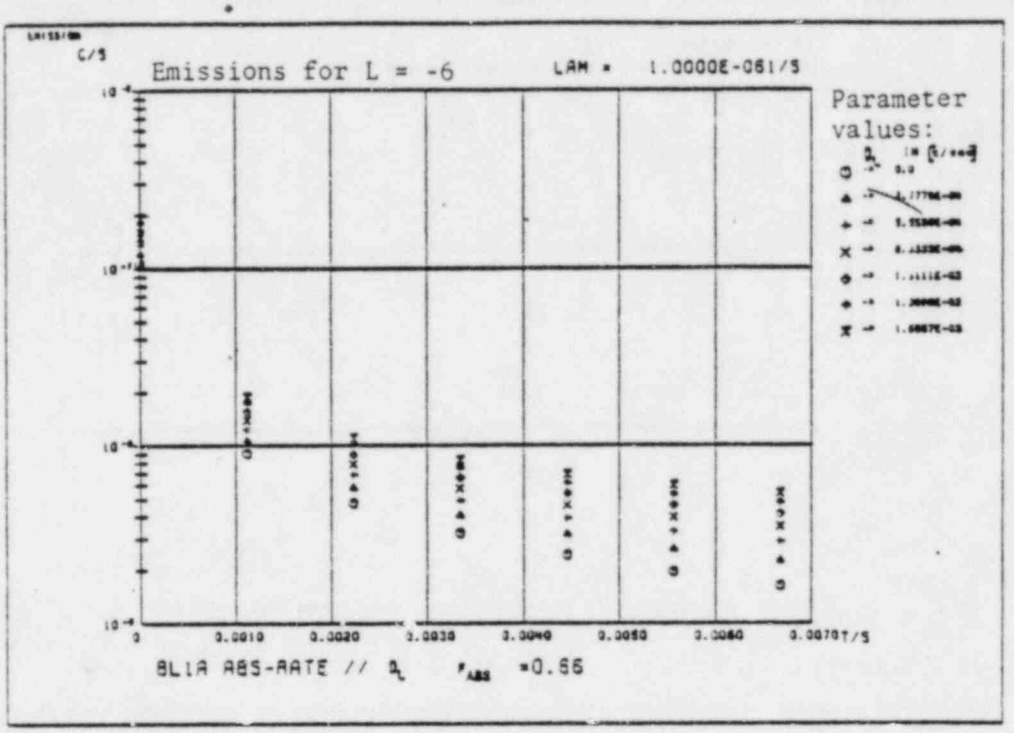
[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:
DFAL = 1.0000E+04 dimensionless
F (10, 9) = 0.0 IN %

BL1A ABS-RATE

NP (1,10) IN T/S / NP (9,13) IN T/S	NP (1,10)	0.0	2.78E-04	5.56E-04	8.33E-04	1.11E-03	1.39E-03	1.67E-03
0.0	-1.07E-07	-1.21E-07	-1.36E-07	-1.50E-07	-1.64E-07	-1.78E-07	-1.92E-07
1.11E-03	4.62E-09	5.25E-09	5.89E-09	6.53E-09	7.17E-09	7.80E-09	8.44E-09
2.22E-03	2.36E-09	2.68E-09	3.01E-09	3.34E-09	3.66E-09	3.99E-09	4.31E-09
3.33E-03	1.58E-09	1.80E-09	2.02E-09	2.24E-09	2.46E-09	2.68E-09	2.89E-09
4.44E-03	1.19E-09	1.36E-09	1.52E-09	1.68E-09	1.85E-09	2.01E-09	2.18E-09
5.56E-03	9.56E-10	1.09E-09	1.22E-09	1.35E-09	1.48E-09	1.61E-09	1.74E-09
6.67E-03	7.98E-10	9.07E-10	1.02E-09	1.13E-09	1.23E-09	1.34E-09	1.45E-09

Figure and Table 4.1.7. Block 1 - Blowdown Demineralization Emission Rates as a Function of Blowdown Rate ABS for Variable Steam Emission Rate D_L and DF_{ABS} = 0.0%.



[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:

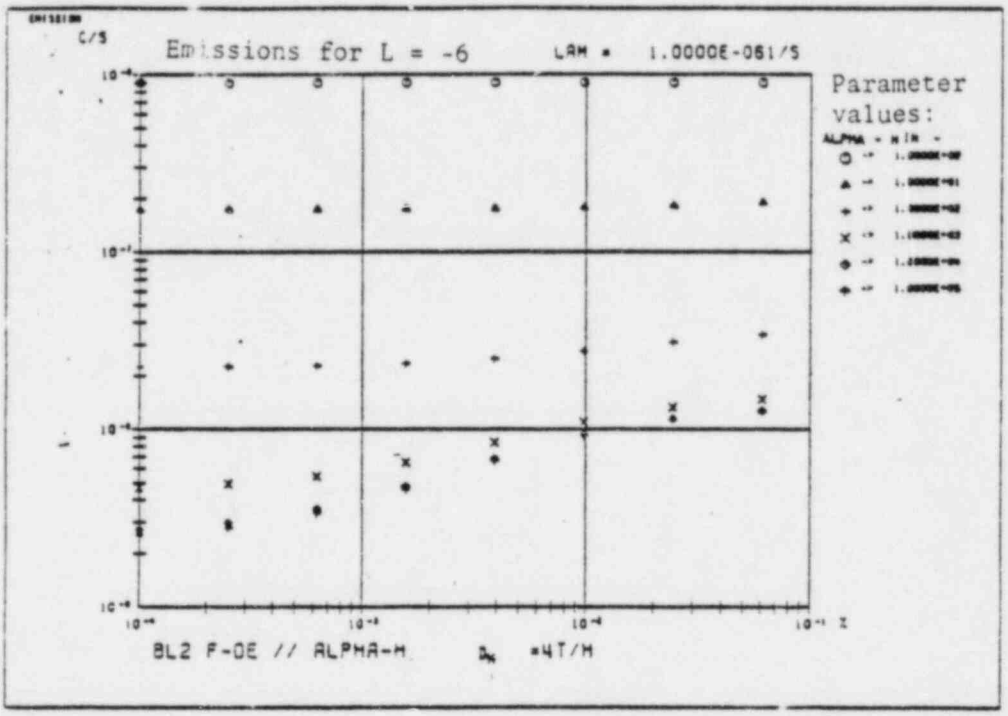
DFAL = 1.0000E+04 dimensionless
F (10, 9) = 6.6660E-01 IN %

BLIA ABS-RATE

| SP (1,10) IN T/S
| SP (9,13) IN T/S

SP (1,10)	0.0	2.78E-04	5.56E-04	8.33E-04	1.11E-03	1.39E-03	1.67E-03
0.0	-1.07E-07	-1.21E-07	-1.36E-07	-1.50E-07	-1.64E-07	-1.78E-07	-1.92E-07
1.11E-03	9.08E-09	1.07E-08	1.23E-08	1.40E-08	1.56E-08	1.72E-08	1.88E-08
2.22E-03	6.79E-09	5.79E-09	6.84E-09	7.88E-09	8.92E-09	9.97E-09	1.10E-08
3.33E-03	3.21E-09	4.05E-09	4.89E-09	5.73E-09	6.56E-09	7.40E-09	8.24E-09
4.44E-03	2.43E-09	3.16E-09	3.89E-09	4.62E-09	5.35E-09	6.08E-09	6.82E-09
5.56E-03	1.95E-09	2.62E-09	3.28E-09	3.95E-09	4.62E-09	5.28E-09	5.95E-09
6.67E-03	1.63E-09	2.25E-09	2.88E-09	3.50E-09	4.12E-09	4.74E-09	5.37E-09

Figure and Table 4.1.8. Block 1 - Blowdown Demineralization Emission Rates as a Function of Blowdown Rate ABS for Variable Steam Emission Rate D_L and DF_{ABS} = 66%.

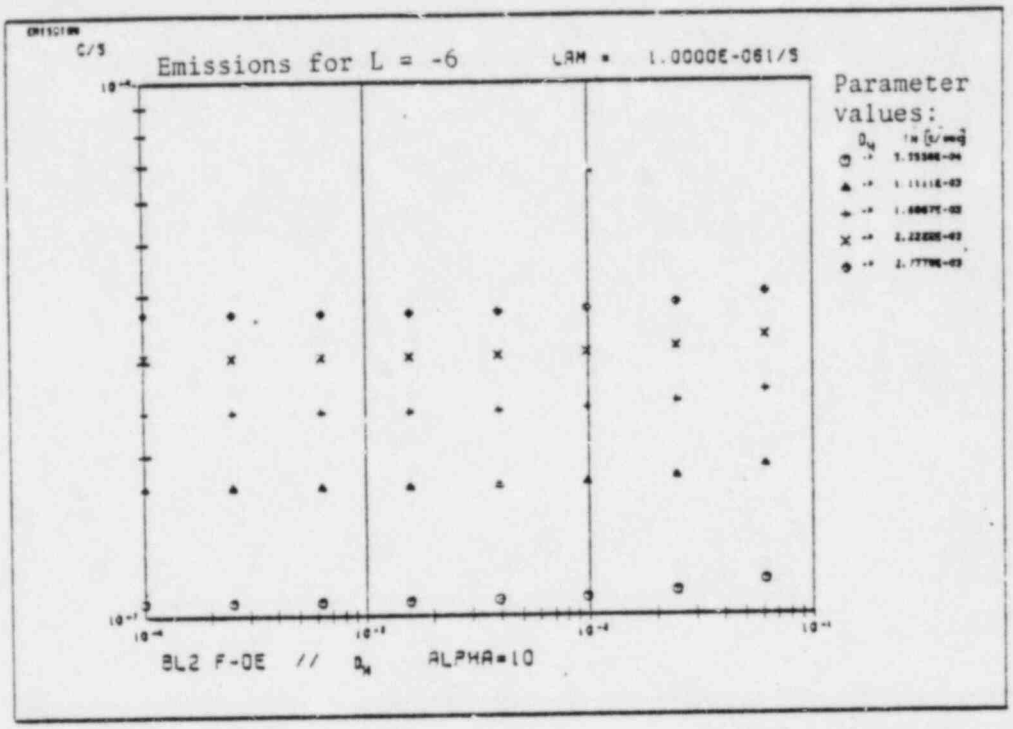


[E- = .0¯]

Emissions in curies per second
For nuclide: L = -6
Parameter list:
BP (3,13) = 1.1111E-03 IN T/S
BL2 F-OE

F (1, 3) IN % / ALPHA H	1.00E+00	1.00E+01	1.00E+02	1.00E+03	1.00E+04	1.00E+05
1.00E-04	9.02E-07	1.74E-07	2.23E-08	4.69E-09	2.73E-09	2.52E-09
2.50E-04	9.02E-07	1.74E-07	2.24E-08	4.91E-09	2.97E-09	2.77E-09
6.25E-04	9.02E-07	1.74E-07	2.27E-08	5.41E-09	3.54E-09	3.35E-09
1.56E-03	9.02E-07	1.75E-07	2.34E-08	6.48E-09	4.73E-09	4.56E-09
3.91E-03	9.02E-07	1.75E-07	2.48E-08	8.40E-09	6.77E-09	6.61E-09
9.77E-03	9.02E-07	1.77E-07	2.74E-08	1.10E-08	9.29E-09	9.13E-09
2.44E-02	9.02E-07	1.82E-07	3.09E-08	1.33E-08	1.14E-08	1.12E-08
6.10E-02	9.02E-07	1.91E-07	3.41E-08	1.47E-08	1.27E-08	1.25E-08

Figure and Table 4.2.2. Block 2 - High Pressure Section Emission Rate as a Function of Residual Moisture F_{SG} for Variable Phase Distribution Factor α_H with D_H = 4 t/hr.



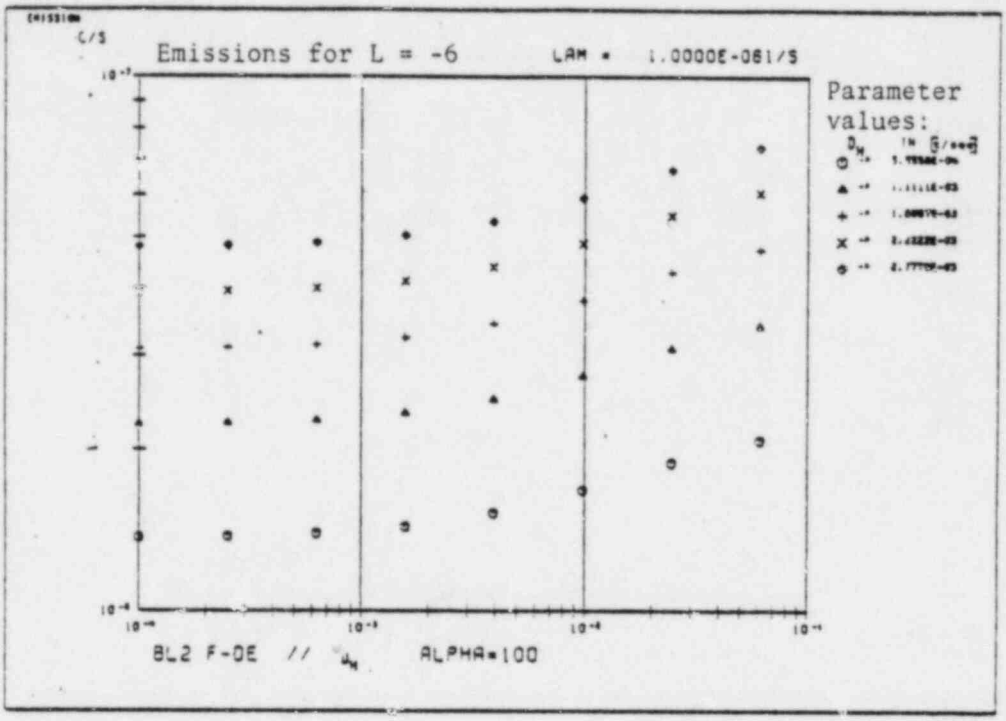
[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list: AL = 1.0000E+01 dimensionless

BL2 F-0E

F (1, 3) \ SF (3,13)	5.56E-04	1.11E-03	1.67E-03	2.22E-03	2.78E-03
1.00E-04	1.06E-07	1.74E-07	2.40E-07	3.05E-07	3.68E-07
2.50E-04	1.06E-07	1.74E-07	2.40E-07	3.05E-07	3.68E-07
6.25E-04	1.06E-07	1.74E-07	2.41E-07	3.05E-07	3.69E-07
1.56E-03	1.06E-07	1.75E-07	2.41E-07	3.06E-07	3.69E-07
3.91E-03	1.07E-07	1.75E-07	2.42E-07	3.08E-07	3.71E-07
9.77E-03	1.08E-07	1.77E-07	2.45E-07	3.11E-07	3.75E-07
2.44E-02	1.11E-07	1.82E-07	2.51E-07	3.19E-07	3.85E-07
6.10E-02	1.16E-07	1.91E-07	2.63E-07	3.34E-07	4.02E-07

Figure and Table 4.2.3. Block 2 - High Pressure Section Emission Rate as a Function of Residual Moisture F_{SG} for Variable Steam Emission Rate D_H and α_H = 10.



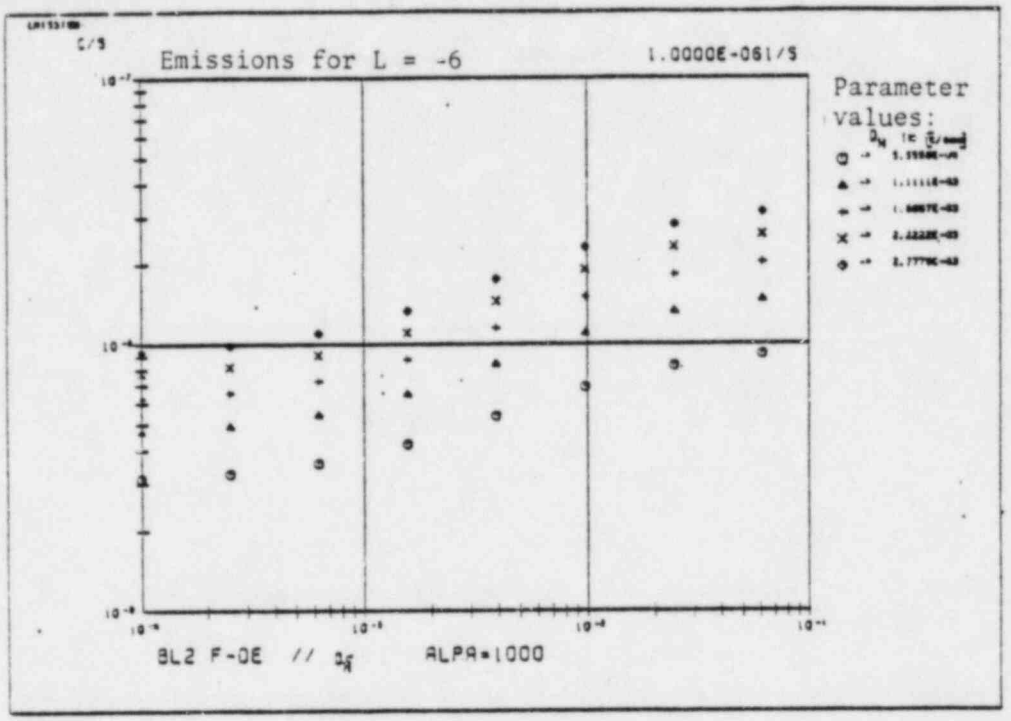
Emissions in curies per second [E- = 10^-]
For nuclide: L = -6
Parameter list:

ALPHA = 1.0000E+02 dimensionless

BL2 F-0E

F (I, 3) \ DH (J,13) IN Ci/S	5.56E-04	1.11E-03	1.67E-03	2.22E-03	2.78E-03
1.00E-04	1.37E-08	2.23E-08	3.09E-08	3.94E-08	4.80E-08
2.50E-04	1.38E-08	2.24E-08	3.11E-08	3.97E-08	4.82E-08
6.25E-04	1.40E-08	2.27E-08	3.15E-08	4.02E-08	4.89E-08
1.56E-03	1.44E-08	2.34E-08	3.24E-08	4.14E-08	5.03E-08
3.91E-03	1.52E-08	2.48E-08	3.44E-08	4.40E-08	5.35E-08
9.77E-03	1.68E-08	2.74E-08	3.81E-08	4.86E-08	5.92E-08
2.44E-02	1.89E-08	3.09E-08	4.29E-08	5.49E-08	6.68E-08
6.10E-02	2.08E-08	3.41E-08	4.74E-08	6.06E-08	7.37E-08

Figure and Table 4.2.4. Block 2 - High Pressure Section Emission Rate as a Function of Residual Moisture F_{SG} for Variable Steam Emission Rate D_H and α_H = 100.

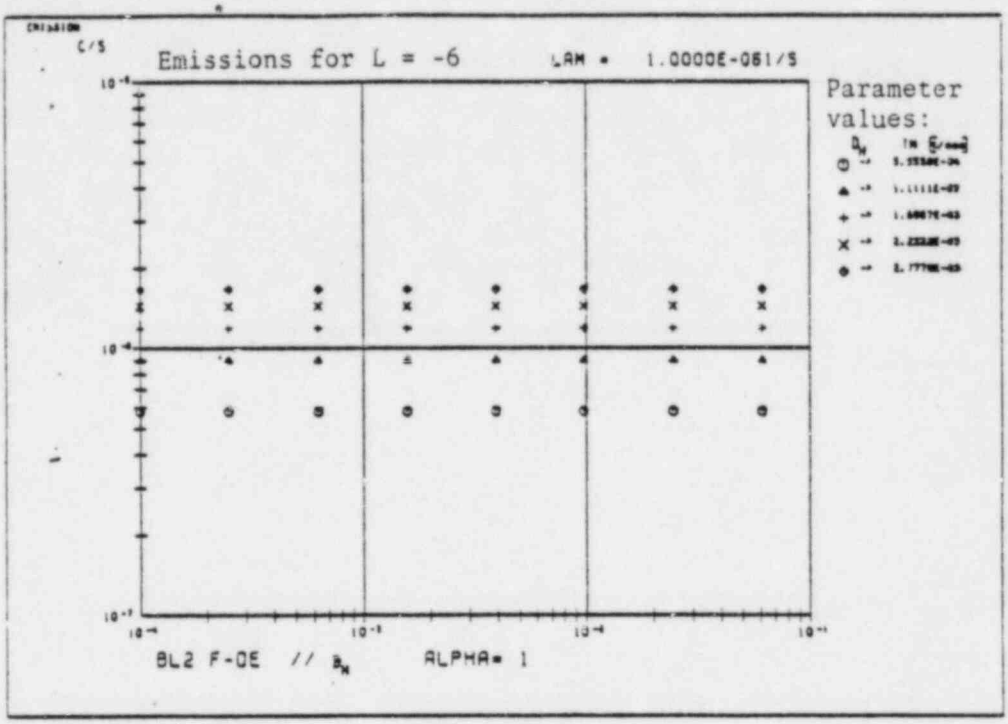


[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:
AL H = 1.0000E+03 dimensionless
BL2 F-0E

F (1, 3) \ DF (3,13) IN T/S	5.56E-04	1.11E-03	1.67E-03	2.22E-03	2.78E-03
1.00E-04	3.13E-09	4.69E-09	6.26E-09	7.82E-09	9.38E-09
2.50E-04	3.26E-09	4.91E-09	6.56E-09	8.21E-09	9.85E-09
6.25E-04	3.56E-09	5.41E-09	7.26E-09	9.11E-09	1.10E-08
1.56E-03	4.20E-09	6.48E-09	8.76E-09	1.10E-08	1.33E-08
3.91E-03	5.36E-09	8.40E-09	1.15E-08	1.45E-08	1.75E-08
9.77E-03	6.88E-09	1.10E-08	1.50E-08	1.91E-08	2.31E-08
2.44E-02	8.27E-09	1.33E-08	1.82E-08	2.32E-08	2.82E-08
6.10E-02	9.14E-09	1.47E-08	2.03E-08	2.58E-08	3.14E-08

Figure and Table 4.2.5. Block 2 - High Pressure Section Emission Rate as a Function of Residual Moisture F_{SG} for Variable Steam Emission Rate D_H and α_H = 1000.



[E- = 10^{-}]

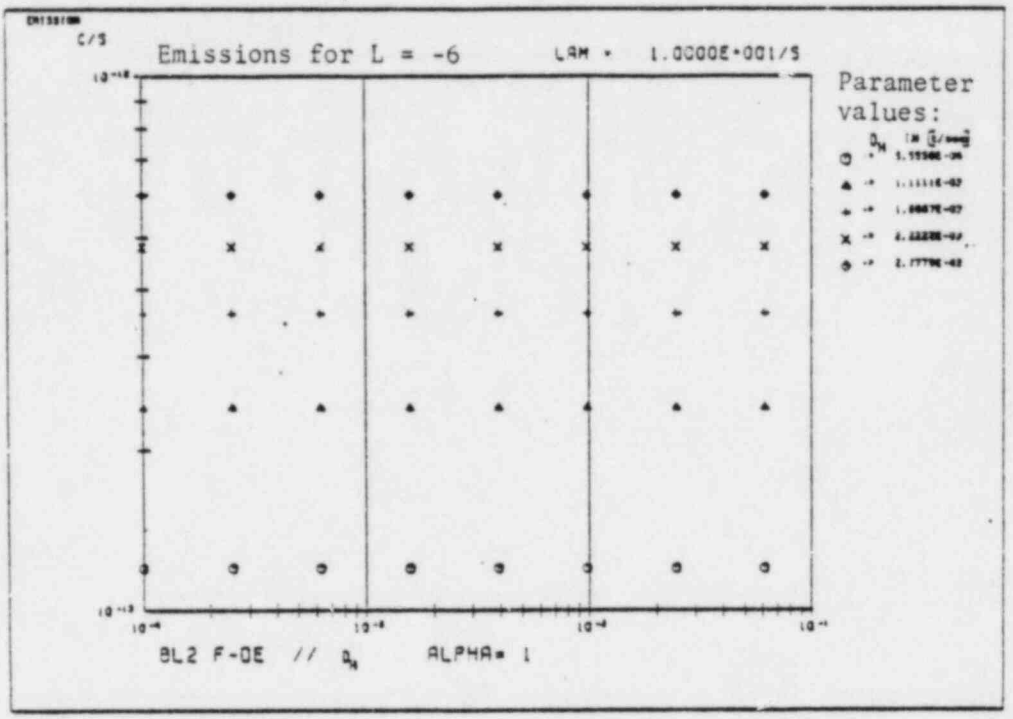
Emissions in curies per second
For nuclide: L = -6
Parameter list:

AL E = 1.0000E+00 dimensionless

BL2 F-0E

F (1, 3) IN % / BP (3,13) IN T/S	5.56E-04	1.11E-03	1.67E-03	2.22E-03	2.78E-03
1.00E-04	5.79E-07	9.02E-07	1.19E-06	1.44E-06	1.66E-06
2.50E-04	5.79E-07	9.02E-07	1.19E-06	1.44E-06	1.66E-06
6.25E-04	5.79E-07	9.02E-07	1.19E-06	1.44E-06	1.66E-06
1.56E-03	5.79E-07	9.02E-07	1.19E-06	1.44E-06	1.66E-06
3.91E-03	5.79E-07	9.02E-07	1.19E-06	1.44E-06	1.66E-06
9.77E-03	5.79E-07	9.02E-07	1.19E-06	1.44E-06	1.66E-06
2.44E-02	5.79E-07	9.02E-07	1.19E-06	1.44E-06	1.66E-06
6.10E-02	5.79E-07	9.02E-07	1.19E-06	1.44E-06	1.66E-06

Figure and Table 4.2.6. Block 2 - High Pressure Section Emission Rate as a Function of Residual Moisture F_{SG} for Variable Steam Emission Rate D_H and $\alpha_H = 1$ and $\lambda = 10^{-6}$ sec^{-1}.

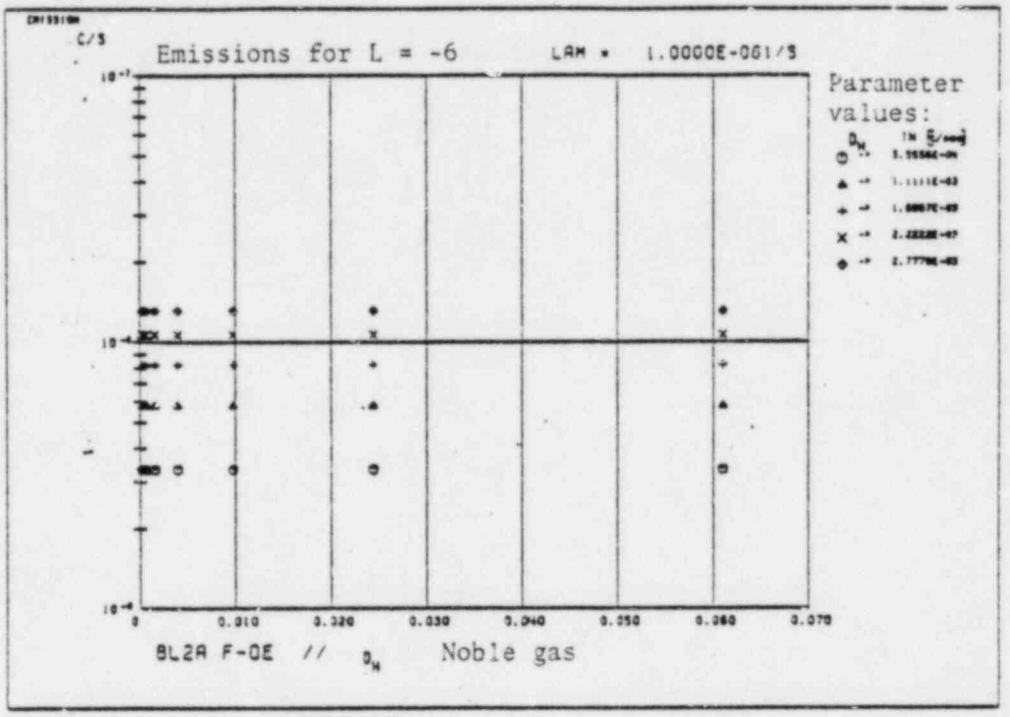


[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:
AL E = 1.0000E+00 dimensionless
BL2 F-0E

F (1, 3) IN % / HP (3,13) IN T/S	5.56E-04	1.11E-03	1.67E-03	2.22E-03	2.78E-03
1.00E-04	1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
2.50E-04	1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
6.25E-04	1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
1.56E-03	1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
3.91E-03	1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
9.77E-03	1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
2.44E-02	1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
6.10E-02	1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13

Figure and Table 4.2.7. Block 2 - High Pressure Range Emission Rate as a Function of Residual Moisture F_{SG} for Variable Steam Emission Rate D_H and $\alpha_H = 1$ and $\lambda = 1$ sec^{-1}.



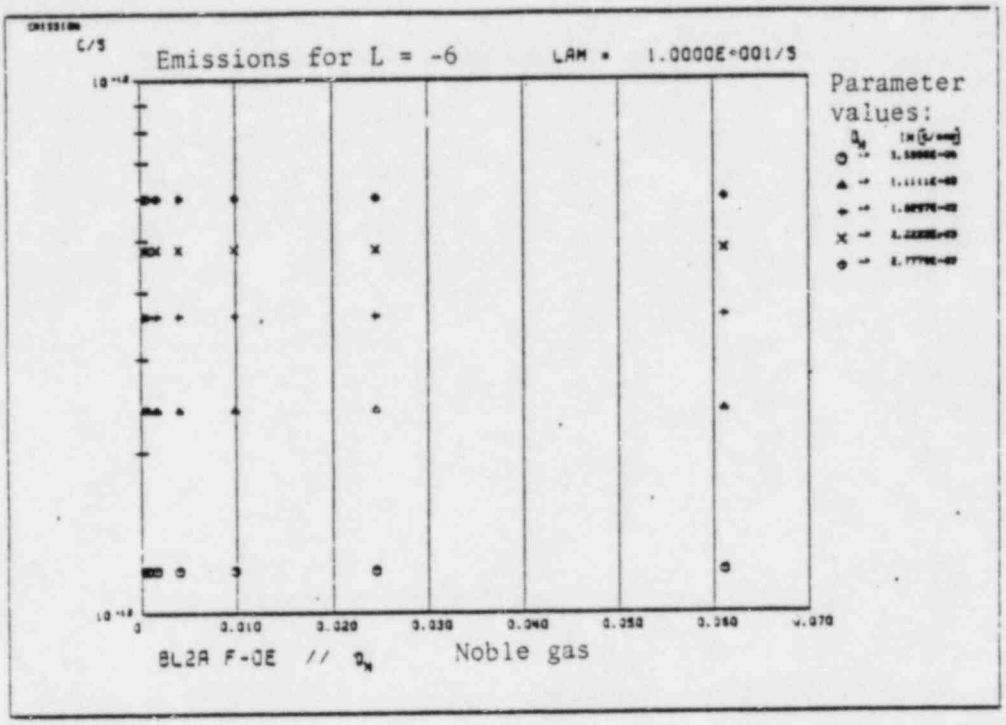
[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:
BL2A F-0E

Noble gas

F (1, 3) IN % / SF (3, 13) IN T/S	5.56E-04	1.11E-03	1.67E-03	2.22E-03	2.78E-03
1.00E-04	3.32E-09	5.76E-09	8.20E-09	1.06E-08	1.31E-08
2.50E-04	3.32E-09	5.76E-09	8.20E-09	1.06E-08	1.31E-08
6.25E-04	3.32E-09	5.76E-09	8.20E-09	1.06E-08	1.31E-08
1.56E-03	3.32E-09	5.76E-09	8.20E-09	1.06E-08	1.31E-08
3.91E-03	3.32E-09	5.76E-09	8.20E-09	1.06E-08	1.31E-08
9.77E-03	3.32E-09	5.76E-09	8.20E-09	1.06E-08	1.31E-08
2.44E-02	3.32E-09	5.76E-09	8.20E-09	1.06E-08	1.31E-08
6.10E-02	3.32E-09	5.76E-09	8.20E-09	1.06E-08	1.31E-08

Figure and Table 4.2.8. Block 2 - High Pressure Section Emission Rate as a Function of Residual Moisture F_{SG} for Variable Steam Emission Rate D_H for Noble Gases with $\lambda = 10^{-6}$ sec^{-1}.

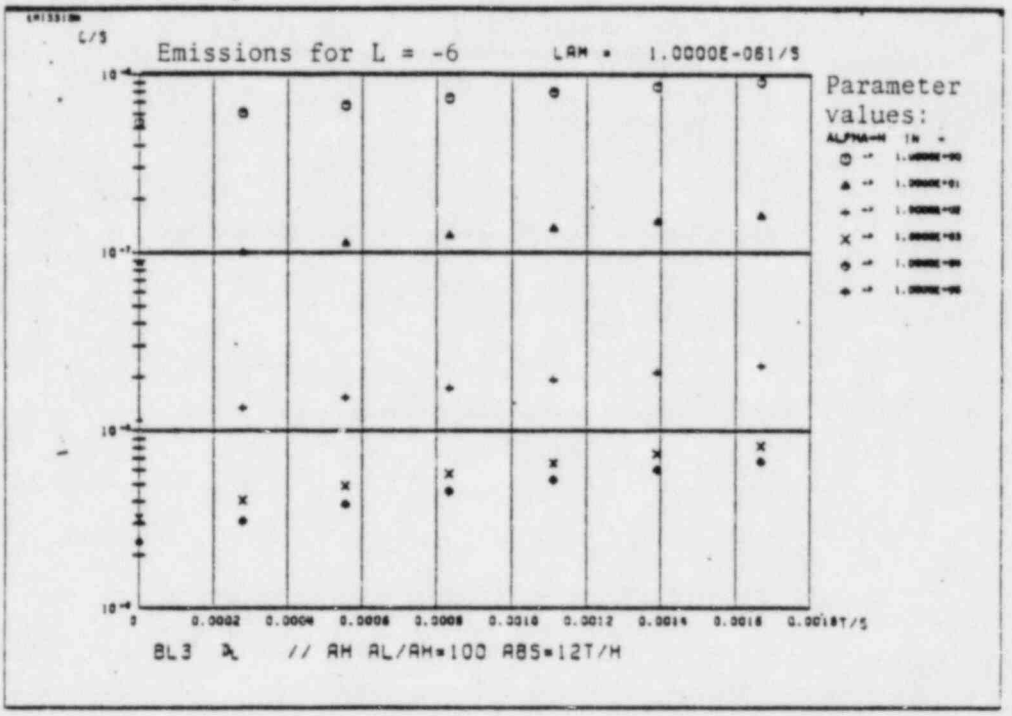


[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:
BL2A F-0E Noble gas

F (1, 3)	FP (3,13) IN T/S	5.56E-04	1.11E-03	1.67E-03	2.22E-03	2.78E-03
1.00E-04		1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
2.50E-04		1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
6.25E-04		1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
1.56E-03		1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
3.91E-03		1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
9.77E-03		1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
2.44E-02		1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13
6.10E-02		1.20E-13	2.40E-13	3.60E-13	4.80E-13	6.00E-13

Figure and Table 4.2.9. Block 2 - High Pressure Section Emission Rate as a Function of Residual Moisture F_{SG} for Variable Steam Emission Rate D_H for Noble Gas with $\lambda = 1$ sec^{-1}.



[E- = 10^{-}]

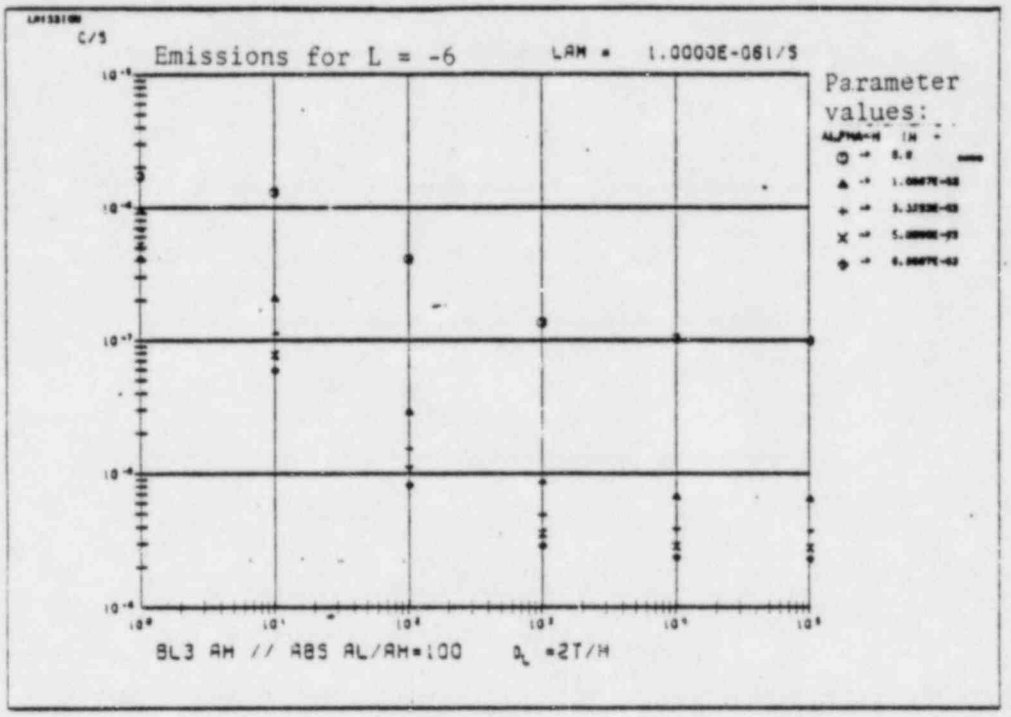
Emissions in curies per second
For nuclide: L = -6
Parameter list:

AL 4 = 1.0000E+02 dimensionless
NP (1,10) = 3.3333E-03 IN T/S

BL3 .

NP (9,13) IN T/S / AL N	1.00E+00	1.00E+01	1.00E+02	1.00E+03	1.00E+04	1.00E+05
NP (9,13)						
0.0	5.38E-07	8.79E-08	1.14E-08	3.21E-09	2.39E-09	2.31E-09
2.78E-04	6.05E-07	1.00E-07	1.33E-08	4.05E-09	3.11E-09	3.02E-09
5.56E-04	6.70E-07	1.12E-07	1.53E-08	4.89E-09	3.84E-09	3.73E-09
8.33E-04	7.33E-07	1.25E-07	1.73E-08	5.73E-09	4.56E-09	4.45E-09
1.11E-03	7.95E-07	1.37E-07	1.92E-08	6.56E-09	5.29E-09	5.16E-09
1.39E-03	8.56E-07	1.49E-07	2.12E-08	7.40E-09	6.01E-09	5.88E-09
1.67E-03	9.14E-07	1.61E-07	2.31E-08	8.24E-09	6.74E-09	6.59E-09

Figure and Table 4.3.2. Block 3 - Low Pressure Section Emission Rates as a Function of Steam Emission Rate D_L for Variable α_H.



[E- = 10^{-}]

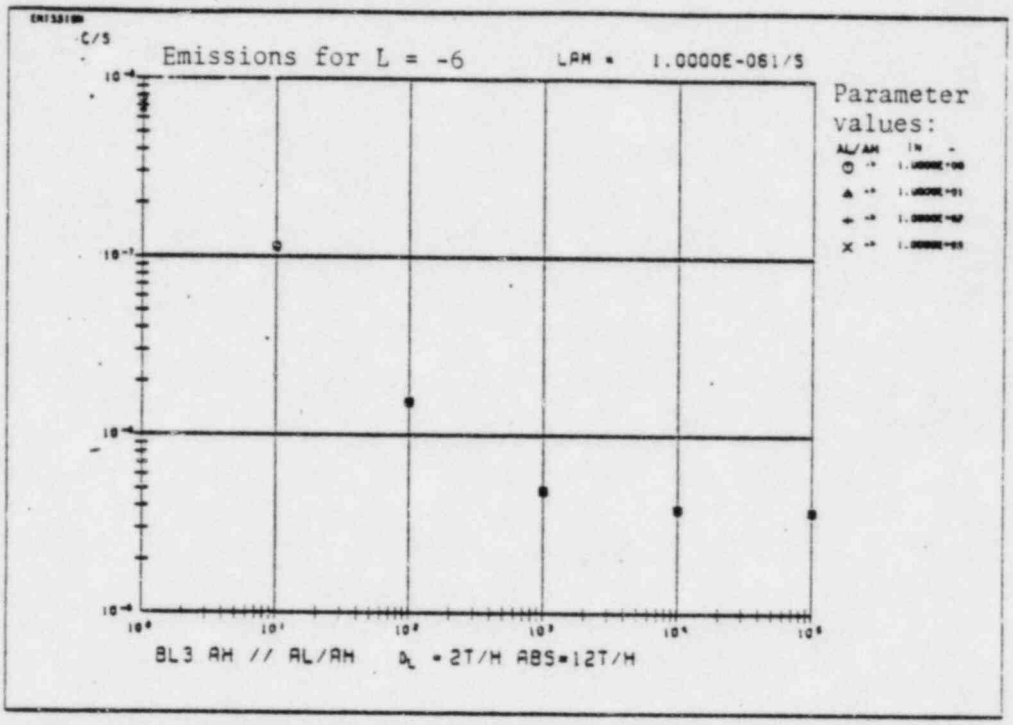
Emissions in curies per second
For nuclide: L = -6
Parameter list:

SP (9,13) = 5.5556E-04 IN T/S
AL L/AK = 1.0000E+02 dimensionless

BL3 AH // ABS AL/AH=100

AL H	AL H SP (1,10) IN T/S dimensionless 0.0	1.67E-03	3.33E-03	5.00E-03	6.67E-03
1.00E+00	1.72E-06	9.64E-07	6.70E-07	5.13E-07	4.16E-07
1.00E+01	1.30E-06	2.06E-07	1.12E-07	7.74E-08	5.90E-08
1.00E+02	4.08E-07	2.88E-08	1.53E-08	1.06E-08	8.17E-09
1.00E+03	-1.36E-07	8.73E-09	4.89E-09	3.55E-09	2.88E-09
1.00E+04	-1.03E-07	6.69E-09	3.84E-09	2.85E-09	2.34E-09
1.00E+05	-9.93E-08	6.49E-09	3.73E-09	2.78E-09	2.29E-09

Figure and Table 4.3.5. Block 3 - Low Pressure Section Emission Rates as a Function of Phase Distribution Factor α_H for Variable Blowdown Rate ABS.



[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:

SP (9,11) = 5.5556E-04 IN T/S
SP (1,10) = 3.3333E-03 IN T/S

BL3 AH // AL/AH

AL H / AL L/AH	1.00E+00	1.00E+01	1.00E+02	1.00E+03
1.00E+00	7.72E-07	[illegible]	6.70E-07	6.79E-07
1.00E+01	1.14E-07	[illegible]	1.12E-07	1.12E-07
1.00E+02	1.53E-08	[illegible]	1.53E-08	1.53E-08
1.00E+03	4.89E-09	[illegible]	4.89E-09	4.89E-09
1.00E+04	3.84E-09	3.84E-09	3.84E-09	3.84E-09
1.00E+05	3.73E-09	3.73E-09	3.73E-09	3.73E-09

dimensionless

Figure and Table 4.3.4. Block 3 - Low Pressure Section Emission Rates as a Function of Phase Distribution Factor α_H for Variable Ratio α_L/α_H.

Emissions in curies per second
For nuclide: L = -6

[E- = 10^{-}]

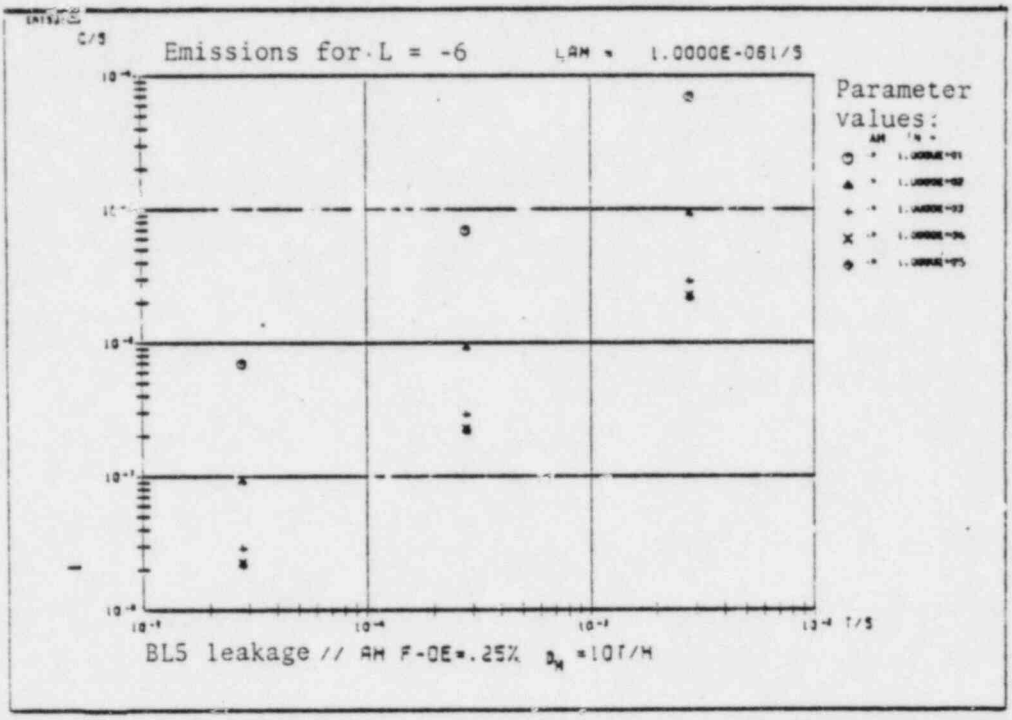
Primary conc. CS/T	Nuclide L= 1	L= 2	L= 3	L= 4	L= 6	L= 8	L=10	L=12
1.00E-00	8.92E-24	1.13E-19	1.35E-18	1.26E-17	6.03E-17	7.14E-17	7.14E-17	7.14E-17
1.00E-07	8.92E-23	1.13E-18	1.35E-17	1.26E-16	6.03E-16	7.14E-16	7.14E-16	7.14E-16
1.00E-06	8.92E-22	1.13E-17	1.35E-16	1.26E-15	6.03E-15	7.14E-15	7.14E-15	7.14E-15
1.00E-05	8.92E-21	1.13E-16	1.35E-15	1.26E-14	6.03E-14	7.14E-14	7.14E-14	7.14E-14
1.00E-04	8.92E-20	1.13E-15	1.35E-14	1.26E-13	6.03E-13	7.14E-13	7.14E-13	7.14E-13
1.00E-03	8.92E-19	1.13E-14	1.35E-13	1.26E-12	6.03E-12	7.14E-12	7.14E-12	7.14E-12
1.00E-02	8.92E-18	1.13E-13	1.35E-12	1.26E-11	6.03E-11	7.14E-11	7.14E-11	7.14E-11
1.00E-01	8.92E-17	1.13E-12	1.35E-11	1.26E-10	6.03E-10	7.14E-10	7.14E-10	7.14E-10
1.00E+00	8.92E-16	1.13E-11	1.35E-10	1.26E-09	6.03E-09	7.14E-09	7.14E-09	7.14E-09
1.00E+01	8.92E-15	1.13E-10	1.35E-09	1.26E-08	6.03E-08	7.14E-08	7.14E-08	7.14E-08

Table 4.4.2. Block 4 - Standard Computation Emission Rates as a Function of Primary Concentration A for Varied Decay Constants λ for Salts.

Emissions in curies per second
Parameter list:
BL4 primary concentration // nuclide gases [E- = 10^{-}]

Primary concen.	Nuclide L= 1	L= 2	L= 3	L= 4	L= 6	L= 8	L=10	L=12
1.00E-08	2.16E-21	1.40E-17	3.50E-17	4.97E-17	5.28E-17	5.28E-17	5.28E-17	5.28E-17
1.00E-07	2.16E-20	1.40E-16	3.50E-16	4.97E-16	5.28E-16	5.28E-16	5.28E-16	5.28E-16
1.00E-06	2.16E-19	1.40E-15	3.50E-15	4.97E-15	5.28E-15	5.28E-15	5.28E-15	5.28E-15
1.00E-05	2.16E-18	1.40E-14	3.50E-14	4.97E-14	5.28E-14	5.28E-14	5.28E-14	5.28E-14
1.00E-04	2.16E-17	1.40E-13	3.50E-13	4.97E-13	5.28E-13	5.28E-13	5.28E-13	5.28E-13
1.00E-03	2.16E-16	1.40E-12	3.50E-12	4.97E-12	5.28E-12	5.28E-12	5.28E-12	5.28E-12
1.00E-02	2.16E-15	1.40E-11	3.50E-11	4.97E-11	5.28E-11	5.28E-11	5.28E-11	5.28E-11
1.00E-01	2.16E-14	1.40E-10	3.50E-10	4.97E-10	5.28E-10	5.28E-10	5.28E-10	5.28E-10
1.00E+00	2.16E-13	1.40E-09	3.50E-09	4.97E-09	5.28E-09	5.28E-09	5.28E-09	5.28E-09
1.00E+01	2.16E-12	1.40E-08	3.50E-08	4.97E-08	5.28E-08	5.28E-07	5.28E-08	5.28E-08

Figure 4.4.3. Block 4 - Standard Computation Emission Rates as a Function of Primary Concentration A for Varied Decay Constants λ for Gases.



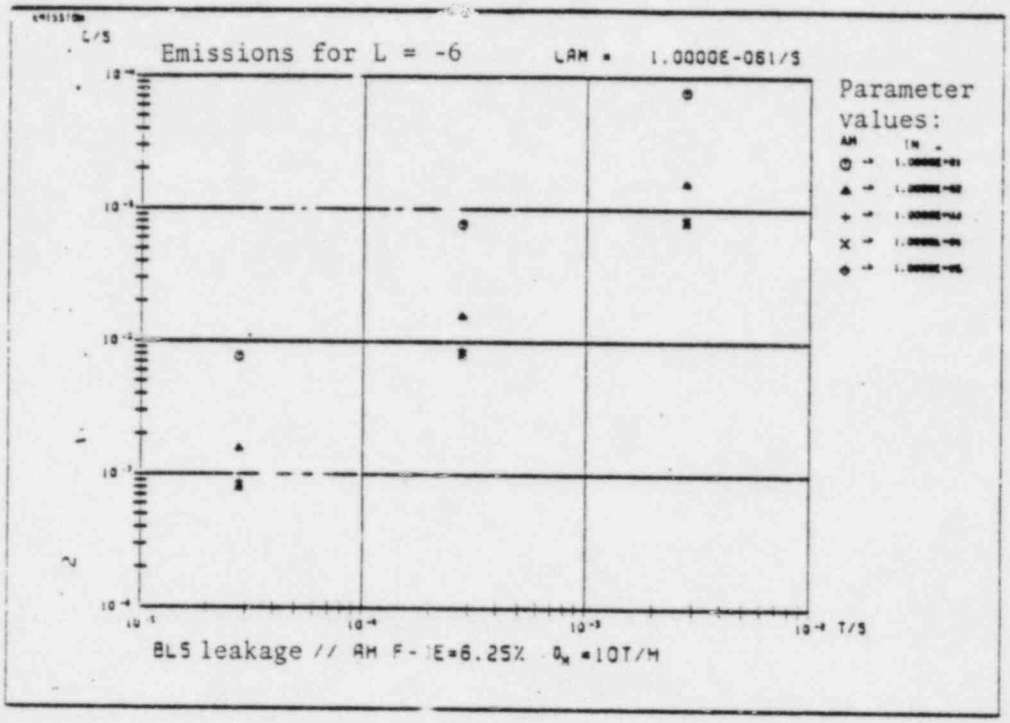
[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:

F (1, 3) = 2.5000E-03 [X %
RF (3,13) = 2.7778E-03 [X C/S
BL5 leakage// AH F-OE=.25% 1

Leak	Leak / AL H 1.00E+01	[X T/S 1.00E+02	1.00E+03	1.00E+04	1.00E+05
2.78E-05	6.91E-07	9.21E-08	2.88E-08	2.25E-08	2.18E-08
2.78E-04	6.91E-06	9.21E-07	2.88E-07	2.25E-07	2.18E-07
2.78E-03	6.91E-05	9.21E-06	2.88E-06	2.25E-06	2.18E-06

Figure and Table 4.5.2. Block 5 - Extremal Computation Emission Rates as a Function of Leakage for Varied α_H for F_{SG} = 0.25%.



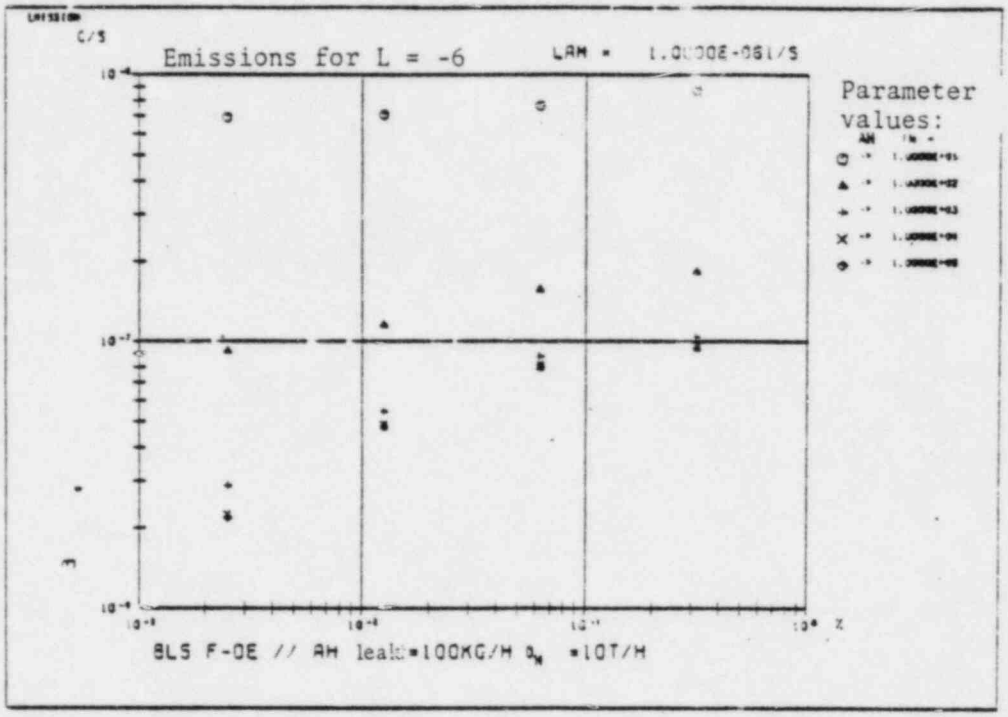
[E- = 10^{-}]

Emissions in curies per second.
For nuclide:
Parameter list:

P (1, 3)= 6.2500E-02 IN %
HP (3,13)= 2.7778E-03 IN T/S
BLS leakage // AH F-02=6.25%

Leak | AL S	IN T/S				
Leak	1.00E+01	1.00E+02	1.00E+03	1.00E+04	1.00E+05
2.78E-05	7.56E-07	1.57E-07	8.80E-08	8.09E-08	8.02E-08
2.78E-04	7.56E-06	1.57E-06	8.80E-07	8.09E-07	8.02E-07
2.78E-03	7.56E-05	1.57E-05	8.80E-06	8.09E-06	8.02E-06

Figure and Table 4.5.3. Block 5 - Extremal Computation Emission Rates as a Function of Leakage for Varied α_H for P_{SG} = 6.25%.

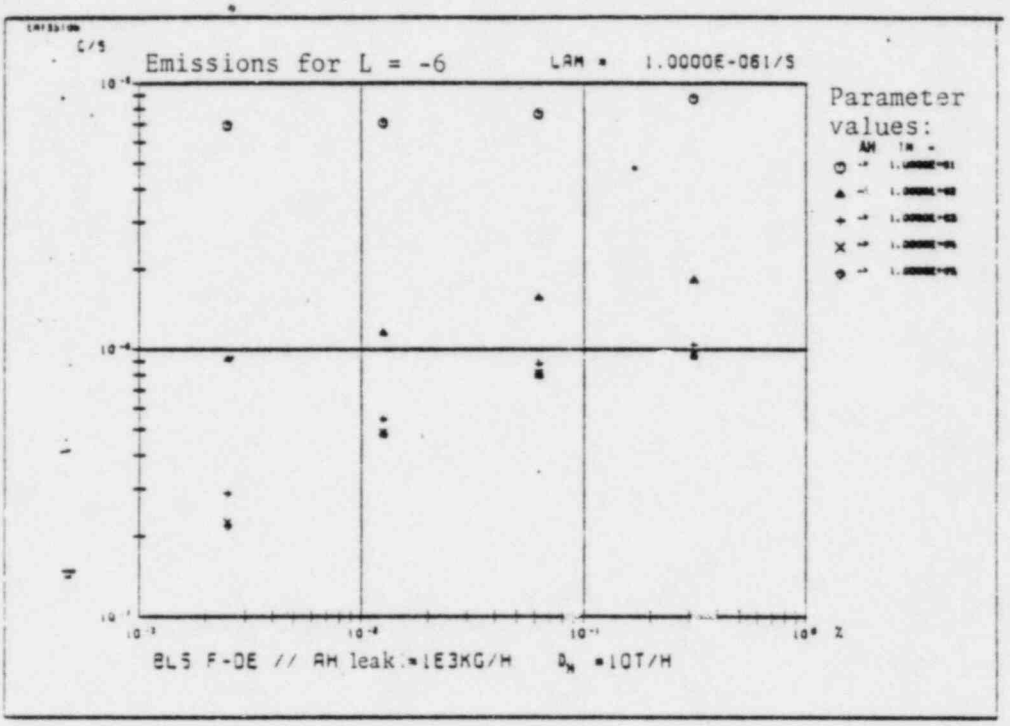


[E- = 10⁻]

Emissions in curies per second
For nuclide:
Parameter list:
Leak = 2.7778E-05 IN T/S
XP (3, 13) = 2.7778E-03 IN T/S
BLS F-0E // AH leak =100KG/H ABS=10T/H 3

F (1, 3) IN %					
F (1, 3) | ALL E	1.00E+01	1.00E+02	1.00E+03	1.00E+04	1.00E+05
2.50E-03	6.91E-07	9.21E-08	2.88E-08	2.25E-08	2.18E-08
1.25E-02	7.07E-07	1.15E-07	5.46E-08	4.86E-08	4.80E-08
6.25E-02	7.66E-07	1.57E-07	8.30E-08	8.09E-08	8.02E-08
3.12E-01	8.74E-07	1.83E-07	1.04E-07	9.56E-08	9.48E-08

Figure and Table 4.5.4. Block 5 - Extremal Computation Emission Rates as a Function of Residual Moisture F_{SG} for Variable α_H and a Leak Rate L = 0.1 t/hr.



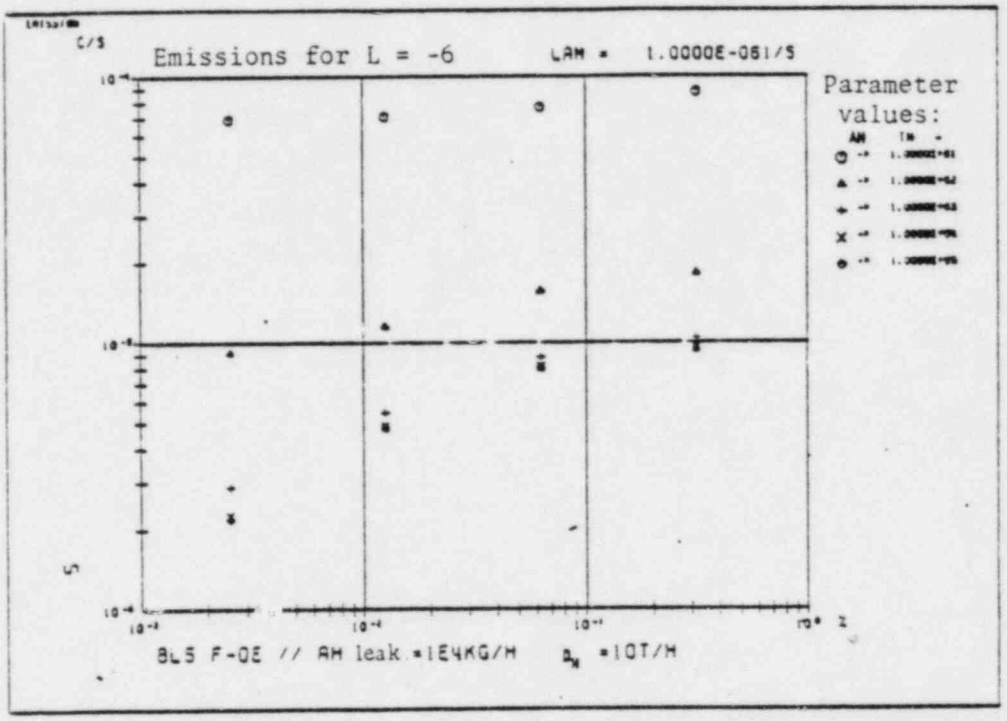
[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:

leak = 2.7778E-04 IN T/S
SP (3,13) = 2.7778E-03 IN T/S
BLS F-DE // AH leak = 1E3KG/H

F (1, 3) IN % / AL H	1.00E+01	1.00E+02	1.00E+03	1.00E+04	1.00E+05
2.50E-03	6.91E-06	9.21E-07	2.88E-07	2.25E-07	2.18E-07
1.25E-02	7.07E-06	1.15E-06	5.46E-07	4.86E-07	4.80E-07
6.25E-02	7.66E-06	1.57E-06	8.80E-07	8.09E-07	8.02E-07
3.12E-01	8.74E-06	1.83E-06	1.04E-06	9.56E-07	9.48E-07

Figure and Table 4.5.5. Block 5 - Extremal Computation Emission Rates as a Function of Residual Moisture F_{SG} for Variable α_H and a Leak Rate L = 1 t/hr.

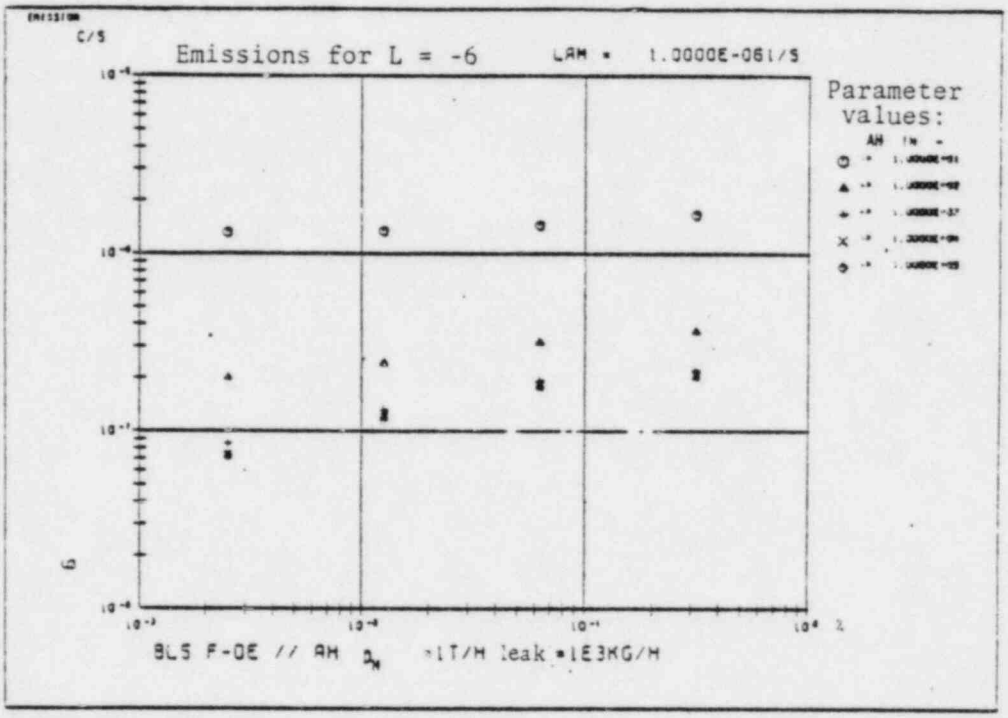


[E- = 10^{-}]

Emissions in curies per second
For nuclide: L = -6
Parameter list:
leak = 2.7778E-03 IN T/S
SP (3,13) = 2.7778E-03 IN T/S
BLS F-0E // AM leak =1E4KG/H 5

F (1, 3) IN % |AL %	1.00E+01	1.00E+02	1.00E+03	1.00E+04	1.00E+05
2.50E-03	6.91E-05	9.21E-06	2.88E-06	2.25E-06	2.18E-06
1.25E-02	7.07E-05	1.15E-05	5.46E-06	4.86E-06	4.80E-06
6.25E-02	7.66E-05	1.57E-05	8.80E-06	8.09E-06	8.02E-06
3.12E-01	8.74E-05	1.83E-05	1.04E-05	9.56E-06	9.48E-06

Figure and Table 4.5.6. Block 5 - Extremal Computation Emission Rates as a Function of Residual Moisture F_{SG} for Variable α_H and a Leak Rate of L = 10 t/hr.



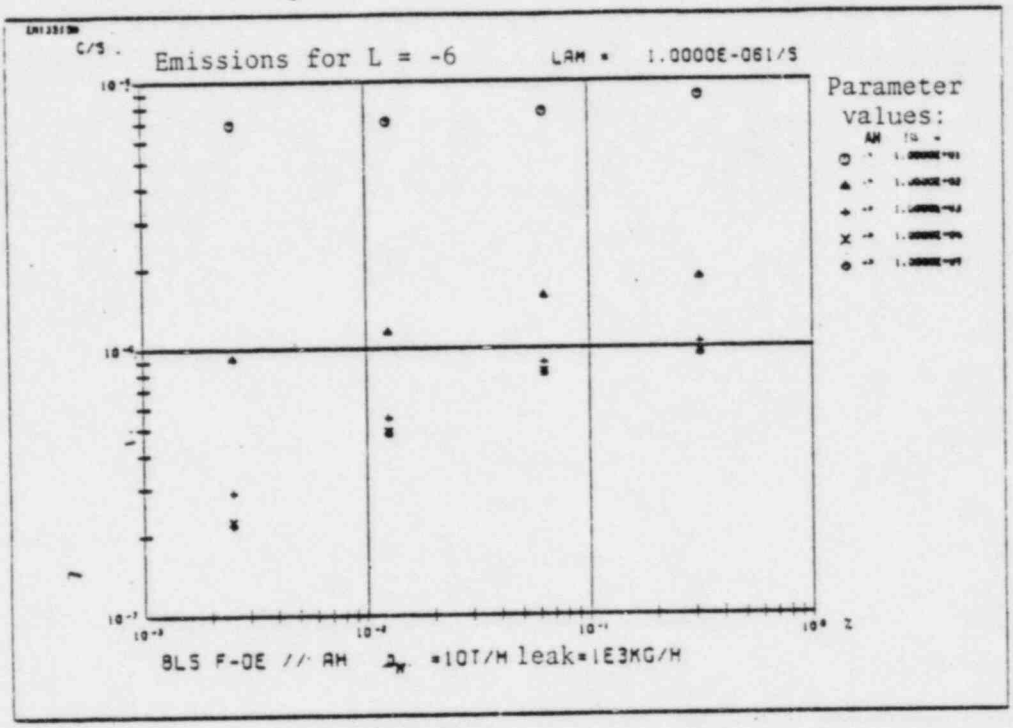
[E- = 10^{-}]

Emissions in curies per second
For nuclide:
Parameter list:

Leak = 2.7778E-04 IN T/S
SP (3,13) = 2.7778E-04 IN T/S
BLS F-0E // AH Leak =1E3KG/H 6

F (1, 3) IN % / AL H	1.00E+01	1.00E+02	1.00E+03	1.00E+04	1.00E+05
2.50E-03	1.31E-06	2.00E-07	8.49E-08	7.34E-08	7.23E-08
1.25E-02	1.34E-06	2.42E-07	1.31E-07	1.21E-07	1.20E-07
6.25E-02	1.45E-06	3.17E-07	1.92E-07	1.79E-07	1.78E-07
3.12E-01	1.65E-06	3.64E-07	2.20E-07	2.06E-07	2.04E-07

Figure and Table 4.5.7. Block 5 - Extremal Computation Emission Rates as a Function of Residual Moisture F_{SG} for Variable α_H and Steam Emission D_H = 1 t/hr.



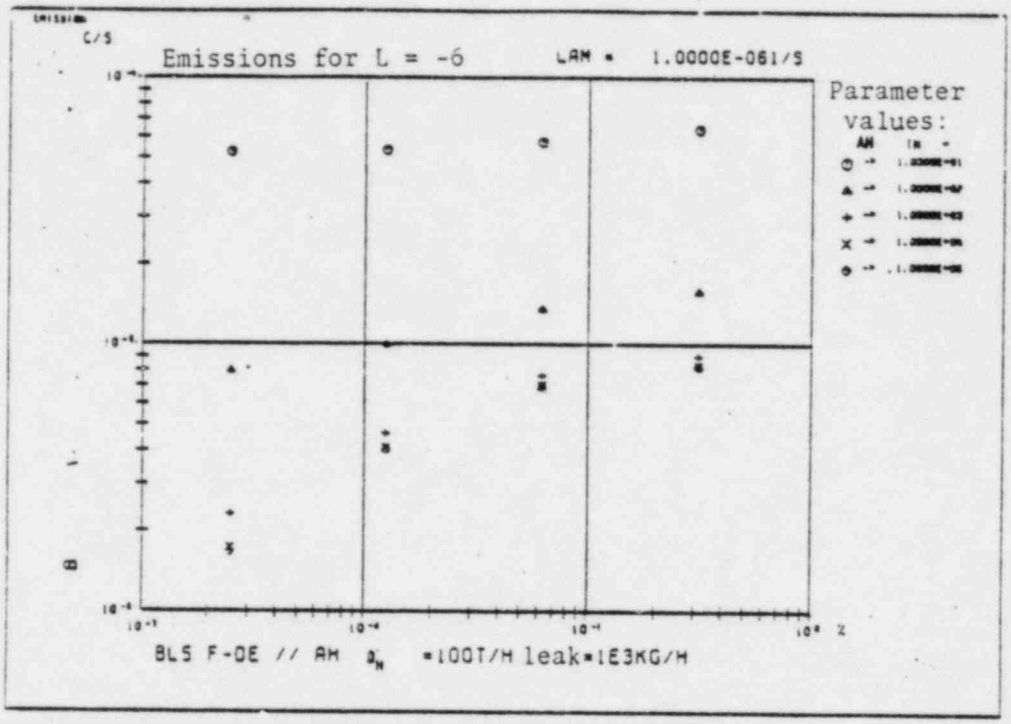
[E- = 10^{-1}]

Emissions in curies per second
For nuclide:
Parameter list:

leak = 2.7778E-04 IN T/S
SP (3,13) = 2.7778E-03 IN T/S
BLS F-0E // AH leak=1E3KG/H 7

F (1, 3) IN % / AL H	1.00E+01	1.00E+02	1.00E+03	1.00E+04	1.00E+05
2.50E-03	6.91E-06	9.21E-07	2.88E-07	2.25E-07	2.18E-07
1.25E-02	7.07E-06	1.15E-06	5.46E-07	4.86E-07	4.80E-07
6.25E-02	7.66E-06	1.57E-06	8.80E-07	8.09E-07	8.02E-07
3.12E-01	8.74E-06	1.83E-06	1.04E-06	9.56E-07	9.48E-07

Figure and Table 4.5.8. Block 5 - Extremal Computation Emission Rates as a Function of Residual Moisture F_{SG} for Variable α_H and Steam Emission D_H = 10 t/hr.

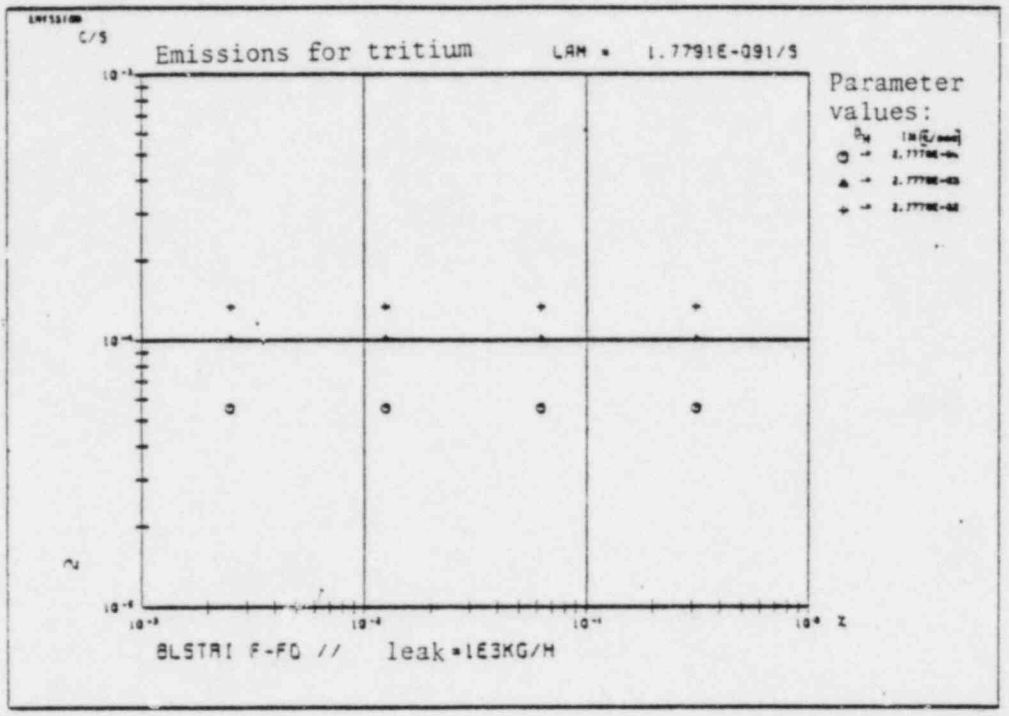


Emissions in curies per second
For nuclide: L = -6 [E- = 10^-]
Parameter list:

leak = 2.7778E-04 IN C/S
SF (3,13) = 2.7778E-02 IN C/S
BLS F-0E // AH leak=1E3KG/H S

F (1, 3) IN % / AL S	1.00E+01	1.00E+02	1.00E+03	1.00E+04	1.00E+05
2.50E-03	5.26E-05	7.93E-06	2.31E-06	1.73E-06	1.67E-06
1.25E-02	5.36E-05	9.97E-06	4.64E-06	4.10E-06	4.05E-06
6.25E-02	5.74E-05	1.36E-05	7.64E-06	7.02E-06	6.96E-06
3.12E-01	6.41E-05	1.58E-05	9.04E-06	8.33E-06	8.26E-06

Figure and Table 4.5.9. Block 5 - Extremal Computation Emission Rates as a Function of Residual Moisture F_{SG} for Variable α_H and Steam Emission D_H = 100 t/hr.



Emissions in curies per second
For nuclide: tritium
Parameter list:

leak = 2.7778E-04 IN I/S
BLSTRI F-FD // DE leak=1E3KG/E 2

F (1, 3) IN S HP (3,13) IN I/S			
F (1, 3)	2.78E-04	2.78E-03	2.78E-02
2.50E-03	5.55E-05	1.01E-04	1.33E-04
1.25E-02	5.55E-05	1.01E-04	1.33E-04
6.25E-02	5.55E-05	1.01E-04	1.33E-04
3.12E-01	5.55E-05	1.01E-04	1.33E-04

Figure and Table 4.5.10. Block 5 - Extremal Computation Emission Rates as a Function of Residual Moisture F_{SG} for Variable D_H.

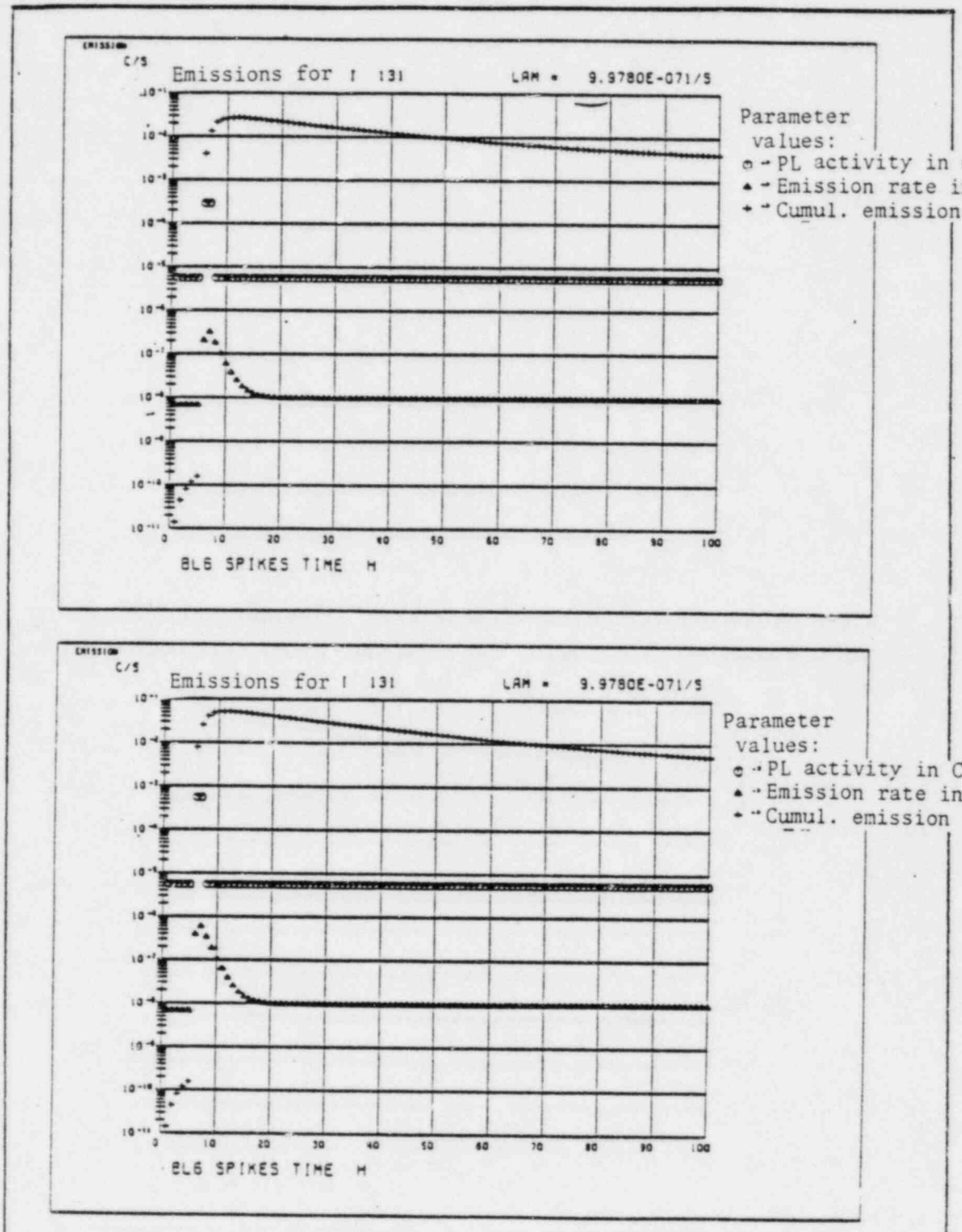


Figure 4.6.2./3. Block 6 - Spike Computation and Load Change Emission Computations for Iodine-131 with Medium Spike Factor (52, Upper Figure) and Maximum Spike Factor (100, Lower Figure) at a Load Change of 100% to 85% in 20 Hrs.

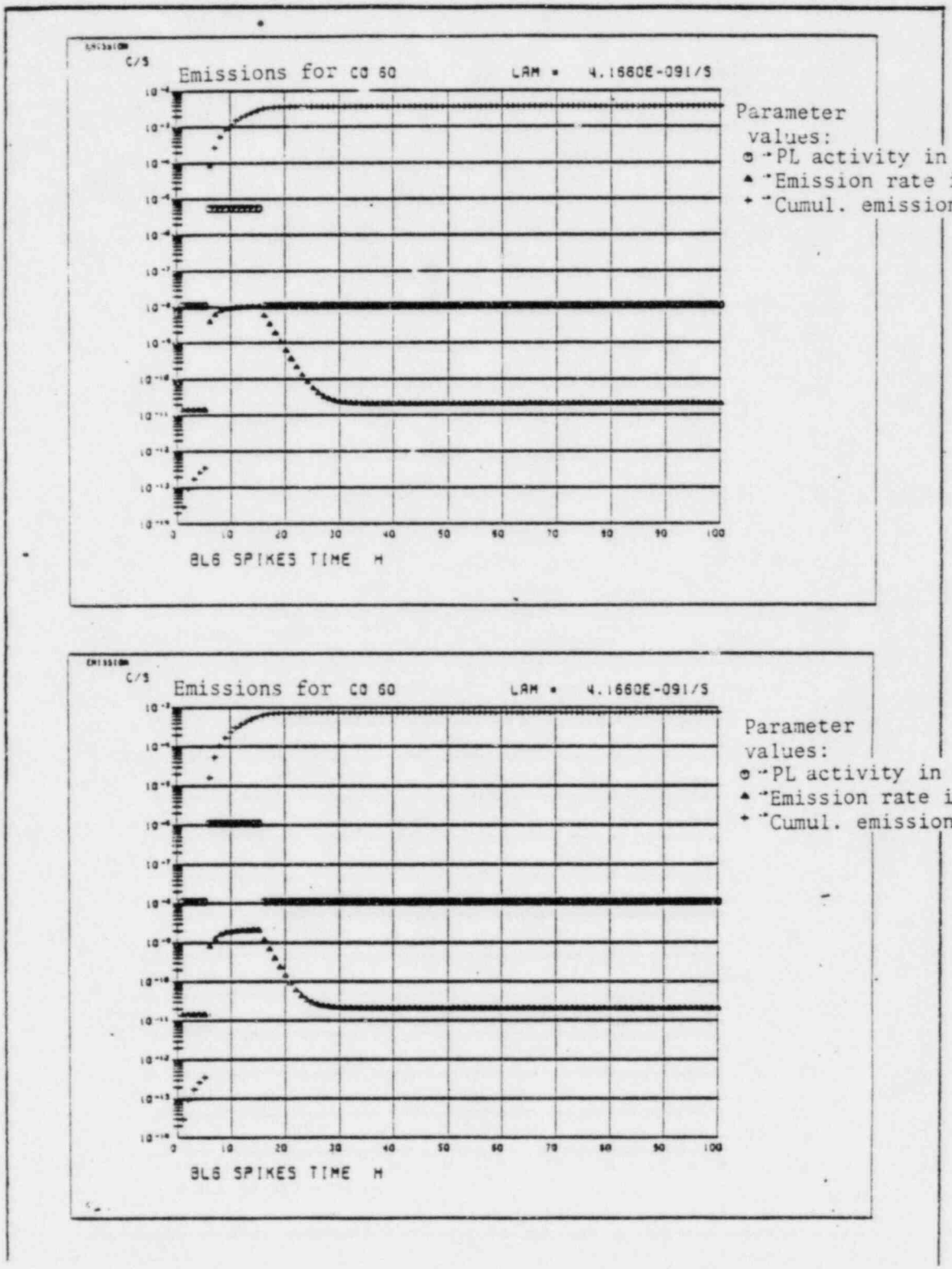


Figure 4.6.4./5. Block 6 - Spike Computation and Load Change Emission Computations for Co-60 with Medium Spike Factor (100, Upper Figure) and Maximum Spike Factor (500, Lower Figure) at a Load Change of 100% to 85% in 20 Hrs.

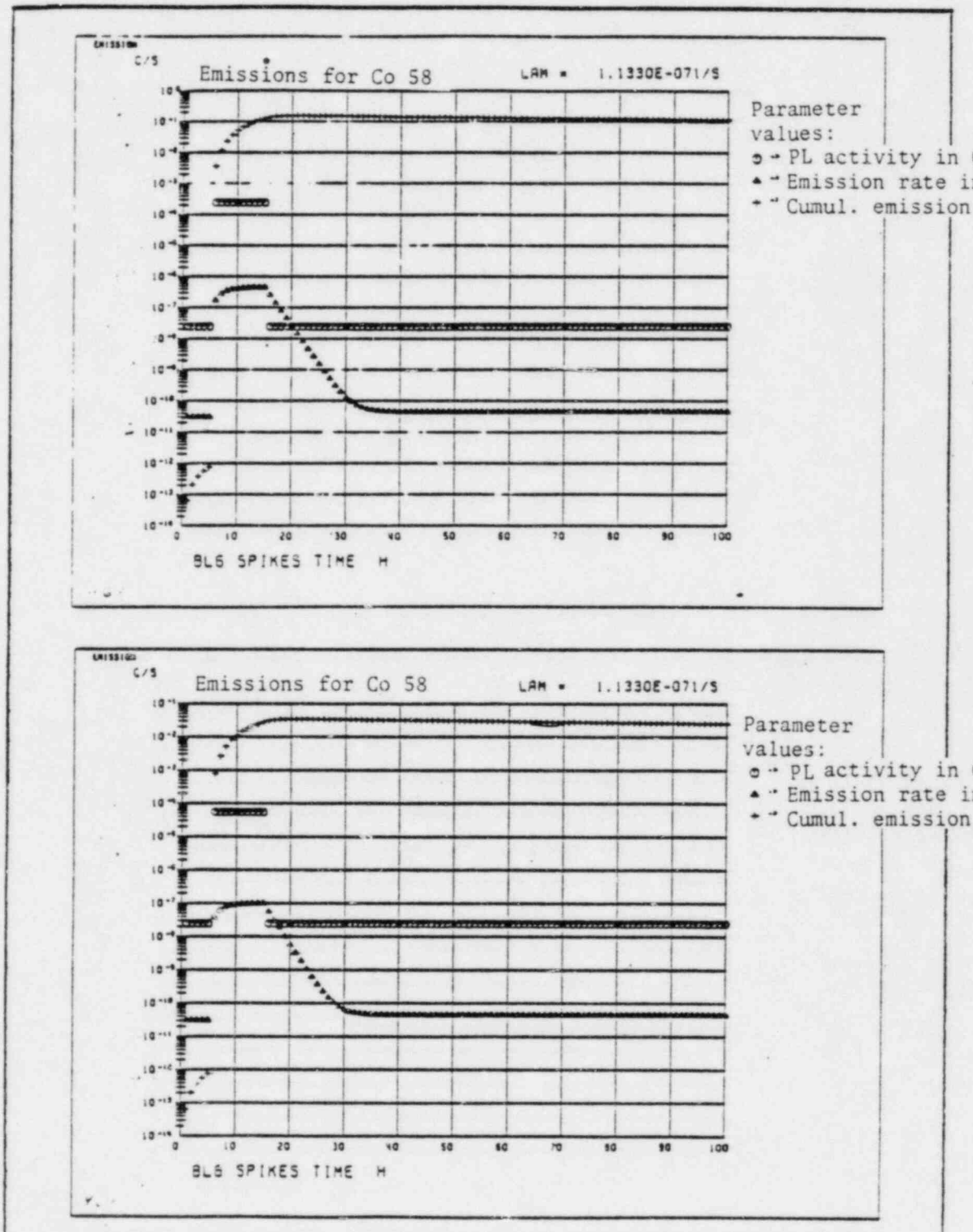


Figure 4.6.6./7. Block 6 - Spike Computation and Load Change Emission Computations for Co-58 with Medium Spike Factor (2.2 10^3, Upper Figure) and Maximum Spike Factor (10^4, Lower Figure) at a Load Change of 100% to 85% in 20 Hrs.

5. Emission Computations for Operational Leaks and Their Evaluation

- Introduction

Specific emission computations with boundary conditions as can occur in normal operation are carried out and evaluated in this section.

Let us refer to the results of the systems analysis of Section 2 for the initial values. Nuclide-specific emissions are evaluated by comparison of emission computations with statutory allowable immission limits.

Allowable dose limits for realistic conditions are already exceeded in steam generator leaks of 25 and 50 kg/hr.

5.1. Emission Computations

Section 2.4 shows that continuous adherence to statutory maximum I 131 live steam concentration is not guaranteed since, on the one hand, false parameters are used in the corresponding computations even today (compare minutes of discussion deadline for Neckarwestheim II, December 1981) (for example pure residual moisture transport of I 131) and, on the other hand, continuous measurement of I 131 main steam concentration during power operation is not stipulated.

Adherence to statutory maximum allowable I 131 main steam concentration is not guaranteed by the N-16 signal of the reactor protection system either. Therefore in emission computations we must take into account that statutory upper I 131 main steam concentration is exceeded. However, it can be assumed that an activity concentration of greater than 10^{-6} Ci/t I 131 in main steam would last only a limited period of time.

The aforementioned indicates that maximum operating leaks in the steam generator cannot be extrapolated from statutory I 131 main steam concentration as is common in official evaluation of nuclear facilities (compare Bibliography 3, 4, as well as Sections 2, 4).

A leak of 25 and 50 kg/hr in a steam generator is assumed in the computation. The other three steam generators are assumed to be perfectly sealed.

Table 5.1.1 which follows lists which nuclide-specific emissions cause a leak of 25 kg/hr in the steam generator. The values derived in Section 2 (Table 2.3.1.2.1) on the basis of measurements without a spike were selected as the primary activities, for the iodine isotope the design values of the KWU as cited were chosen.

In order to select an operating state of maximum realism, the completely intact blowdown demineralization system with a flow rate of 8 [t/hr] per steam generator was assumed. Design residual moisture of 0.25% as well as average phase distribution factors (α values) were adopted in the steam generator. The emissions were computed for a steady-state activity distribution and are listed in Ci/sec.

TABLE 5.1.1. NUCLIDE SPECIFIC POWER HOUSE EMISSIONS IN [CI/SEC]. STEAM GENERATOR LEAK 25 [KG/HR]; STEAM GENERATOR RESIDUAL MOISTURE 0.25%. NUCLIDES WITH $\lambda \leq 10^{-14}$ ARE LISTED AT THE BEGINNING OF THE TABLE.

Nuclide	Emission	Nuclide	Emission
LA 138	2.75D-26	U 238	2.38D-26
ND 144	2.67D-25	NP 236	1.49D-21
SM 147	2.97D-22	NP 237	2.60D-17
SM 148	1.34D-26	PU 242	9.66D-17
GD 152	4.46D-29	CL 36	1.61D-22
TH 232	3.20D-27	RB 87	1.09D-21
U 234	1.26D-17	ZR 93A	2.83D-20
U 235	1.41D-18	ZR 93	1.71D-19
U 236	2.30D-17	PD 107	1.71D-17

TABLE 5.1.1. NUCLIDE-SPECIFIC POWER HOUSE EMISSIONS IN [CI/SEC]. STEAM GENERATOR LEAK 25 [KG/HR]; STEAM GENERATOR RESIDUAL MOISTURE 0.25%. NUCLIDES WITH $\lambda \leq 10^{-14}$ ARE LISTED AT BEGINNING OF TABLE.

Nuclide	Emission	Nuclide	Emission	Nuclide	Emission
H 3	1.92D-06	AS 78	3.38D-11	KR 88	1.27D-09
BE 10	1.23D-16	GE 78	1.76D-13	RB 88	6.86D-10
SI 31	1.11D-14	AS 79	7.36D-12	BR 89	1.98D-14
P 32	1.32D-11	SE 79	1.88D-16	KR 89	7.40D-10
P 33	3.51D-11	SE 79M	4.12D-14	RB 89	6.73D-12
S 35	2.33D-15	AS 81	1.42D-12	SR 89	1.92D-10
CA 45	5.89D-17	KR 81	1.19D-21	KR 90	2.65D-10
SC 46	7.84D-15	SE 81	8.04D-13	RB 90	1.55D-12
SC 47	1.11D-14	SE 81M	7.63D-14	RB 90M	6.61D-13
CR 51	9.71D-11	BR 82	6.68D-12	SR 90A	3.82D-19
MN 54	1.23D-11	BR 82M	6.43D-14	SR 90	1.76D-11
FE 55	1.07D-10	AS 83	5.54D-13	Y 91	2.59D-10
MN 56	1.62D-12	KR 83M	2.36D-10	Y 91M	1.16D-11
CO 58	5.38D-11	SE 83	8.19D-13	KR 91	4.34D-11
FE 59	8.54D-12	SE 83M	5.02D-14	KR 91	1.63D-11
NI 59	2.83D-14	BR 84	1.98D-11	RB 91	6.46D-13
CO 60M	1.33D-12	BR 84M	1.80D-13	SR 91	1.27D-10
CO 60	2.46D-11	SE 84	4.05D-13	Y 92	7.59D-11
CO 61	1.42D-13	BR 85	2.09D-12	RB 92	1.72D-14
NI 63	4.30D-12	KR 85	2.24D-11	SR 92	6.21D-11
CU 64	1.01D-11	KR 85M	1.61D-09	Y 93	1.70D-10
NI 65	1.77D-12	SE 85	3.60E-14	MO 93	1.65D-18
ZN 65	8.35D-13	BR 86	3.90D-13	NB 93M	2.97D-17
GA 72	3.06D-15	BR 86M	1.30D-14	RB 93	2.00D-14
ZN 72	5.99D-11	RB 86	1.16D-12	SR 93	3.54D-12
GA 73	4.17D-15	SE 86	2.65D-14	Y 94	0.0
GE 75	1.08D-14	BR 87	9.52D-13	NB 94	3.94D-20
AS 76	7.80D-13	KR 87	9.06D-10		
AS 77	7.27D-11	SE 87	4.90D-15		
GE 77	1.09D-13	BR 88	2.25D-13		

TABLE 5.1.1. NUCLIDE-SPECIFIC POWER HOUSE EMISSIONS IN [CI/SEC]. STEAM GENERATOR LEAK 25 [KG/HR]; STEAM GENERATOR RESIDUAL MOISTURE 0.25%. NUCLIDES WITH $\lambda \leq 10^{-16}$ ARE LISTED AT BEGINNING OF TABLE.

Nuclide	Emission	Nuclide	Emission	Nuclide	Emission
RB 94	2.17D-15	MO 101	1.40D-11	AG 109M	6.07D-14
SR 94	4.76D-13	NB 101	2.20D-14	PD 109	3.19D-11
Y 95	5.87D-12	TC 101	8.45D-12	PD 109M	1.48D-13
NB 95	3.68D-10	ZR 101	1.29D-18	RH 109	7.53D-14
NB 95M	2.43D-12	MO 102	1.05D-11	RH 109M	2.21D-14
SR 95	1.23D-13	NB 102	1.03D-14	RU 109	2.79D-14
ZR 95	3.67D-13	TC 102	1.90D-14	AG 110	1.97D-14
Y 96	4.31D-14	ZR 102	1.75D-18	AG 110M	1.41D-12
NB 96	3.35D-13	MO 103	8.28D-13	RH 110	9.19D-15
NB 96	9.80D-15	RH 103M	2.19D-11	RU 110	3.22D-14
SR 96	1.10D-16	RU 103	2.65D-10	AG 111	1.13D-11
Y 97	2.54D-16	TC 103	4.03D-13	AG 111M	1.91D-14
NB 97	4.60D-11	MO 104	6.62D-13	PD 111	2.98D-13
NB 97M	4.06D-13	RH 104	2.86D-13	PD 111M	5.40D-14
ZR 97	1.07D-13	TC 104	9.02D-12	RH 111	1.12D-14
Y 98	1.19D-16	MO 105	2.79D-13	AG 112	1.25D-12
NB 98	3.18D-11	RH 105	1.31D-10	PD 112	2.38D-12
ZR 98	2.42D-16	RH 105M	4.85D-14	AG 113	1.23D-12
MO 99	1.52D-12	RU 105	6.43D-11	CD 113M	6.54D-15
MO 99	5.90D-10	TC 105	3.09D-12	PD 113	5.59D-15
TC 99M	3.60D-13	MO 106	2.02D-14	IN 114M	4.06D-16
TC 99M	1.47D-10	RH 106	5.70D-14	PD 114	5.38D-15
TC 99	2.90D-15	RH 106M	1.09D-12	AG 115	3.78D-15
TC 99	5.90D-18	RU 106	8.16D-11	CD 115	1.98D-12
ZR 99	3.74D-18	TC 106	1.19D-13	CD 115M	2.20D-13
TC 100	3.77D-14	RH 107	3.52D-12	IN 115M	6.03D-13
ZR 100	2.98D-17	RU 107	5.46D-13	IN 116	3.50D-14
		TC 107	4.67D-13	CD 117	2.99D-13
		AG 108	8.50D-21	CD 117M	2.21D-13
		RU 108	4.43D-13	IN 117	6.58D-14

TABLE 5.1.1. NUCLIDE-SPECIFIC POWER HOUSE EMISSIONS IN [CI/SEC]. STEAM GENERATOR LEAK 25 [KG/HR], STEAM GENERATOR RESIDUAL MOISTURE 0.25%. NUCLIDES WITH $\lambda \leq 10^{-14}$ ARE LISTED AT BEGINNING OF TABLE.

Nuclide	Emission	Nuclide	Emission	Nuclide	Emission
IN 117M	2.38D-13	SN 127	3.10D-12	J 134M	2.59D-12
SN 117M	4.79D-12	SN 127M	5.40D-14	CS 134	5.44D-11
CD 118	1.86D-13	TE 127	2.59D-11	CS 134M	8.21D-12
IN 119M	3.96D-14	TE 127M	6.41D-12	TE 134	5.05D-11
SN 119M	4.05D-12	J 128	1.04D-12	J 135	1.94D-08
SN 121	1.46D-12	SN 128	3.44D-12	BA 135M	3.75D-14
SN 121M	2.34D-16	J 134	3.18D-09	CS 135M	2.09D-14
SB 122	5.23D-11	SB 129	4.61D-09	CS 135	2.18D-16
SN 123	4.89D-13	TE 129	1.89D-13	TE 135	1.20D-13
SN 123M	3.67D-13	TE 129M	2.45D-11	XE 135	1.30D-08
TE 123	3.96D-28	XE 129M	7.92D-13	XE 135M	3.39D-08
TE 123M	6.41D-15	J 130	2.87D-11	J 136	3.15D-12
SB 124	2.44D-10	J 130M	5.27D-13	J 136M	1.05D-12
SB 125	3.94D-10	SN 130	4.09D-13	CS 136	1.44D-11
SN 125	2.43D-12	J 131	8.35D-09	J 137	7.18D-13
SN 125M	5.25D-14	SB 131	1.28D-09	BA 137M	8.74D-14
TE 125M	9.63D-13	SN 131	7.98D-14	CS 137	5.44D-11
TE 125M	2.81D-13	TE 131	3.19D-11	TE 137	4.62D-15
SB 126	2.96D-11	TE 131M	4.03D-11	XE 137	8.86D-08
SB 126	5.02D-13	XE 131M	1.12D-09	J 138	5.43D-14
SB 127	3.94D-09	J 132	6.35D-09	CS 138	7.38D-10
PU 243	1.12D-11	SN 132	2.71D-14	CS 138M	2.13D-13
PU 243	1.12D-11	TE 132	6.31D-10	XE 138	1.25D-07
AM 244	5.54D-12	J 133	3.20D-08	BA 139	5.17D-11
CM 244	1.71D-13	SB 133	6.85D-11	CS 139	1.25D-11
CM 245	2.30D-17	TE 133	1.12D-11	XE 139	2.40D-08
CM 246	4.68D-18	TE 133M	3.26D-11	BA 140	3.83D-10
TC 108	2.16D-15	XE 133	3.96D-08	CS 140	1.09D-12
		XE 133M	7.24D-09	LA 140	3.35D-10
				XE 140	5.57D-09

TABLE 5.1.1. NUCLIDE-SPECIFIC POWER HOUSE EMISSIONS IN [CI/SEC]. STEAM GENERATOR LEAK 25 [KG/HR]; STEAM GENERATOR RESIDUAL MOISTURE 0.25%. NUCLIDES WITH $\lambda \leq 10^{-14}$ ARE LISTED AT BEGINNING OF TABLE.

Nuclide	Emission	Nuclide	Emission	Nuclide	Emission
BA 141M	1.10D-11	PM 148	6.05D-11	TB 161	2.33D-13
CE 141	2.87D-16	CE 149	2.58D-15	DY 165	2.31D-14
CS 141	2.59D-13	ND 149	1.31D-11	DY 166	1.01D-13
CS 141	2.59D-13	PM 149M	1.40D-11	HO 166	1.84D-14
LA 141	1.14D-10	PM 149	1.15D-10	HO 166M	3.27D-19
BA 142	5.85D-12	PR 149	4.44D-14	TA 182	4.66D-12
CS 142	1.20D-15	PM 150	6.36D-14	W 185	4.28D-13
LA 142	5.07D-11	PR 150	7.66D-15	TH 230	2.82D-22
PR 142	9.68D-12	ND 151	8.92D-13	PA 231	4.46D-16
PR 142M	9.01D-14	PM 151	3.45D-11	U 232	7.43D-20
BA 143	1.09D-13	SM 151	5.35D-14	U 233	1.86D-21
CE 143	2.64D-10	EU 152	9.66D-16	PA 233	2.42D-17
CE 143	2.64D-10	ND 152	5.60D-13	U 237	7.75D-11
LA 143	7.77D-12	PM 152	7.82D-13	PU 237	2.65D-16
PR 143	3.26D-10	PM 152M	8.62D-15	NP 238	3.10D-11
BA 144	3.95D-14	GD 153	1.11D-14	PU 238	2.08D-13
CE 144	2.67D-10	PM 153	1.62D-13	U 239	6.37D-11
CE 144	2.67D-10	EU 154	3.79D-12	NP 239	1.41D-09
PR 144M	3.59D-14	PM 154	2.47D-14	PU 239	2.38D-14
BA 145	6.39D-15	PM 154M	8.67D-15	NP 240	3.60D-13
CE 145	1.01D-12	EU 155	2.23D-12	PU 240	3.64D-14
CE 145	1.01D-12	SM 155	3.31D-13	AM 241	5.65D-15
LA 145	1.18D-13	EU 156	5.37D-11	AM 241	5.65D-15
CE 146	4.03D-12	SM 156	2.79D-12	PU 241	8.17D-12
CE 146	4.03D-12	EU 157	2.18D-12	AM 242M	6.17D-16
CE 146	4.03D-12	SM 157	4.11D-14	AM 242	3.16D-12
LA 146	1.45D-14	EU 158	1.53D-13	AM 242M	6.17D-16
CE 147	1.74D-13	EU 159	2.75D-14	CM 242	2.52D-12
ND 147	1.44D-10	GD 159	8.05D-13	AM 243	1.34D-15
PM 147	2.90D-11	TB 160	2.22D-13	CM 243	3.12D-16

5.2. Estimation of Emissions

To some extent the nuclide-specific emissions differ by more than 10 orders of magnitude. Estimation of all nuclides with regard to their contribution to immission load was not possible within the framework of this study. In this case the following would have to be taken into account in particular:

-- For short-lived nuclides the effects of daughter products formed therefrom;

-- Long-term effects for nuclides with long half-life;

-- The effects of all nuclides which are known simultaneously as essential trace elements or as potential essential trace elements (compare Section 2.3).

In the following, emissions of iodine, cobalt and tritium are examined in greater detail. Using the example of iodine isotope 131, it is shown that a steam generator leak of 25 kg/hr within 20 hrs leads to emissions by means of which the boundary values of the Radiologic Protection Ordinance are far exceeded under realistic and even more dramatically unfavorable boundary conditions (for example rain).

The emission or immission load upon failure of the blowdown demineralization system is the highest.

At this point let us point out that the Baden Technical Supervisory Association assumes for computations within the framework of a guideline to Section 28, paragraph 3 of the Radiologic Protection Ordinance [4] that the blowdown demineralization system operates only 7-14 days annually.

In addition Tables 5.1.1 and also 5.2.1 show that the power house emissions of short-lived iodine isotopes are to some extent considerably greater than I 131 emissions.

This means that this isotope must be considered unconditionally in corresponding expert reports on secondary emissions since otherwise effects are underestimated.

In this regard let us refer to a corresponding evaluation of the Reactor Safety Corporation for the Neckarwestheim II Nuclear Power Plant [5] in which only the load by iodine 131 is considered.

TABLE 5.2.1. POWER HOUSE EMISSIONS OF SELECTED NUCLIDES AND IMMISSION COMPUTATIONS FOR I-131, TIME-DEPENDENT COMPUTATIONS.

1 [illegible] 2 [illegible] / [illegible]	ABS : 8 t/h; Leck: 25 [kg/h]; α_H :800; E in Ci/20h	ABS : 8 t/h; Leck: 50 [kg/h]; α_H :800; E in Ci/20h	ABS : 0; Leck: 25 [kg/h]; α_H :800; E in Ci/20h	ABS : 0; Leck: 50 [kg/h]; α_H :800; E in Ci/20h	ABS : 8 t/h; Leck: 25 [kg/h]; α_H :260; E in Ci/20h	ABS : 8 t/h; Leck: 50 [kg/h]; α_H :260; E in Ci/20h	ABS : 0; Leck: 25 [kg/h]; α_H :260; E in Ci/20h	ABS : 0; Leck: 50 [kg/h]; α_H :260; E in Ci/20h
J-131	0,24	0,48	0.30	0,55	0.54	1,05	0,84	1,60
J 132	0.4	0.8	0.46	0.9	0.64	1.29	0.76	1,50
J 133	1,48	2.9	2.09	4,03	2.27	4.67	3.5	6.75
J 135	1.06	2.11	1.27	2.49	1.07	3.29	2.29	4.49
3 Strahlendosis für den Belastungspfad J 131 Weide-Kuh-Milch-Kleinkind (Schilddrüse) [mrem] Grenzwerte Normalbetrieb 90 mrem, Grenzwerte Störfall 15 000 mrem.								
4 Szenario 1 5 realist. Bedingungen	68	130	100	190	110	210	170	320
6 Szenario 2 7 konservat. Bedingungen	2 300	4 500	3 500	6 600	3 700	7 300	5 800	11 000

Key: 1. Emission conditions E

2. Leak

3. Radiation dose for load pathway of I 131 pasture-cow-milk-child (thyroid) [mrem]. Boundary values normal operation 90 mrem, boundary values accident 15,000 mrem.

4. Scenario 1

5. Realistic conditions

6. Scenario 2

7. Conservative conditions

ABS: Blowdown demineralization system, flow rate per steam generator.

Leak: Leak in one steam generator, the other three steam generators are sealed.

α_H: Phase distribution factor in high pressure section, residual moisture of steam generator = 0.25%.

The scenarios are specified in Table 5.2.2.

The emission data in Table 5.2.1 show the cumulative power house emissions after 20 hrs under different conditions. During the 20 hrs of emission the emission rates rise at varying rates corresponding to the conditions. An emission time of only 20 hrs was assumed although there are many factors which suggest that emissions can last considerably longer than 20 hrs.

Leaks of 25 and 50 kg/hr apply as operational leaks and initially do not deleteriously affect reactor operation. Only when the maximum I 131 main steam concentration is exceeded is intervention into operation planned. However, it must be first established that the main steam concentration has been exceeded. It is easy to imagine that the following procedure lasts longer than 20 hrs:

> Recognition of cause, thereupon measurements in main steam and evaluation of measurements, reaction thereto including notification of authorities.

Computation of Radiation Exposure by I 131 Via Pasture-Cow-Milk-Child Exposure Pathway

A scenario model was used for computation of the expected radiation exposure which takes into account variation of different parameters. The data used are summarized in Table 5.2.2. Data for the air → milk transfer, milk consumption rate and dose factor were derived from the evaluation of Bleck-Neuhaus [6].

TABLE 5.2.2. PARAMETERS OF SCENARIOS FOR EXPOSURE PATHWAY I-131[3] PASTURE-COW-MILK-CHILD (THYROID)

Scenario no.	Atmospheric propagation	Transfer factor[2] milk/air	Milk[2] consumption	Dose factor
1 (realistic conditions)	Avg. at[1] constant wind direction (= 1,5·10⁻⁴ s/m³	High (F = 10 000 m³/l)	High (Q = 1 l/d)	High (D/Q$_1$ = 10 mrem/nCi)
2 (unfavorable conditions)	Constant wind direction, low wind speed (= 1,5·10⁻³ s/m³	High (with rain) (F = 60 000 m³/l)	High (Q = 1 l/d)	Very high (½ maximum) (D/Q$_1$ = 30 mrem/nCi)

[1] Average site with 30% wind direction frequency in sector, with constant wind direction.
[2] According to Bleck-Neuhaus, 1981 [6].
[3] According to [5] the chemical form exclusively as I was assumed.

The short-term ratios must especially be taken into account based on short emission time.

Thus, for example, it is more realistic to assume a higher value for the air → milk transfer factor in the case of brief power house roof discharge than use the annual average for this value (compare 7, 8).

The low emission height of the power house roof (approximately 40 m) was taken into account.

Scenario 1 - realistic conditions: Assumes average propagation conditions with a constant wind direction in a 30° sector. The values derived from Bleck-Neuhaus [6] for the other parameters represent realistic data confirmed by measurements which are especially applicable to short-term conditions (as in this case).

The results of this scenario show that the allowable immission limit for iodine 131 (maximum thyroid exposure 90 mrem) can be exceeded with a steam generator leak of 25 kg/hr during full-time operation of the blowdown demineralization system.

Scenario 2 - unfavorable and therefore conservative conditions: Also takes into account special meteorological situation of the stable weather pattern with low wind speed in which case the air → milk transfer is especially high due to rain. This is caused by the higher deposition of I_2 on wet surfaces.

The higher dose factor as well represents measured conditions as is explained in greater detail by Bleck-Neuhaus [6, 7].

Accordingly, scenario 2 describes unfavorable but realistic conditions as must be taken into account at the least for § 45 of the Radiologic Protection Ordinance [9]:

Extract from § 45 of the Radiologic Protection Ordinance:

> .... "This radiation exposure must be taken into account for the least favorable influence sites with regard to all relevant exposure pathways including the food chain;" ....

The results of this scenario show that the maximum immission limit for iodine is exceeded by a factor of 41 for a steam generator leak of 25 kg/hr during full-time operation of the blowdown demineralization system. When the ABS fails and with a steam generator leak of 50 kg/hr, the allowable accident immission limit for iodine 131 of 15,000 mrem is more than 70% reached at 11,000 mrem.

Table 5.2.2 summarizes the initial parameters of the individual scenarios once more.

Figure 5.2.1 graphs the results of immission computations for I 131:

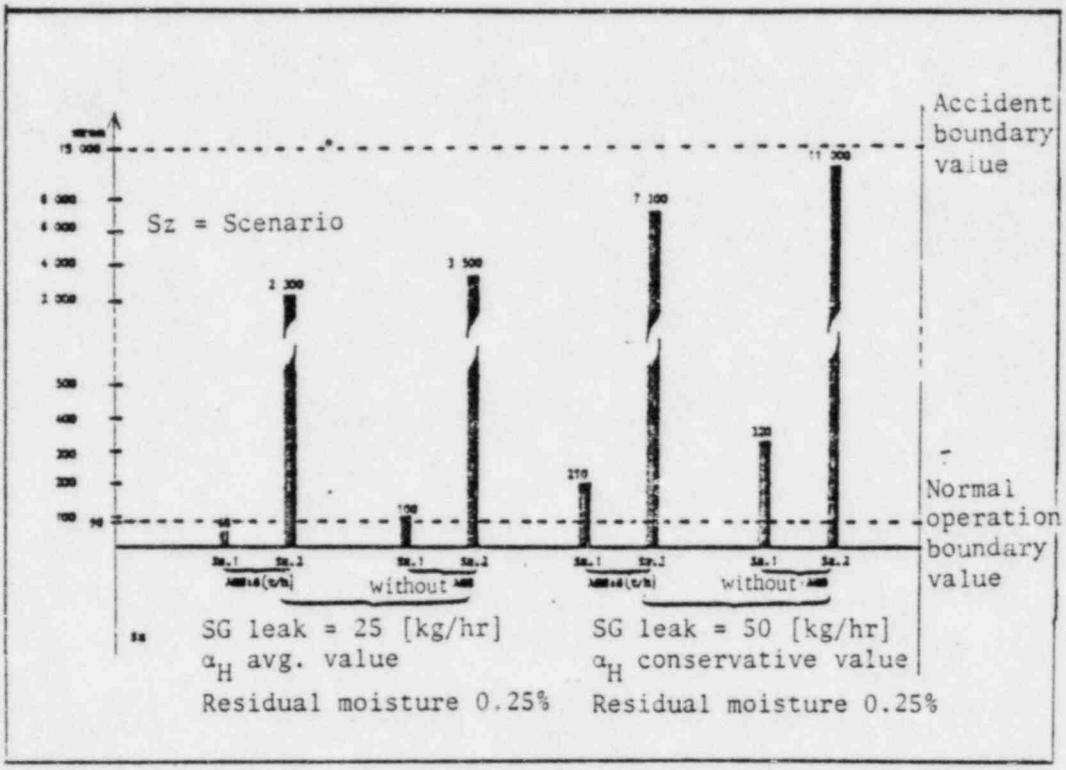


Figure 5.2.1. Radiation Dose for Pasture-Cow-Milk-Child (Thyroid) Exposure Pathway of I 131 [mrem].

In the case of secondary loop emissions, not only iodine emissions, but also other nuclides are of importance, especially if spiking phenomena are considered at the same time.

By way of example let us discuss tritium H3 and cobalt 58 at this point.

Tritium: H3

For Biblis B stack emissions of tritium in 1976 were 3.5 Ci [10]. This corresponds to an average discharge of 8 m Ci in 20 hrs. For a leak of 50 kg/hr in the steam generator the tritium emissions above the power house roof based on the primary loop activity of 0.5 Ci/t are 40 m Ci after 20 hrs. Therefore they exceed the aforementioned averaged stack emissions during this time by a factor of 5.

Cobalt: Co 58

For the Esenshamm Nuclear Power Plant 0.125 Ci/yr emissions were authorized for aerosols. According to the model mixture of the Federal Ministry of the Interior [11] these emissions consist of 25% Co 58. This corresponds to an average discharge of $7.13 \cdot 10^{-5}$ Ci in 20 hrs. Under conservative assumptions (without ABS) $5 \cdot 10^{-5}$ Ci in 20 hrs above the power house roof were determined for a steam generator leak of 50 kg/hr. This value is of the same order of magnitude as the average stack emissions in 20 hrs derived above. Since, however, in the case of Co 58 concentration increases (spikes) of the primary loop concentration involved linearly in the power house emission were found up to 10,000 (compare Table 2.3.1.2.1), this value may be considerably higher.

Under unfavorable conditions the annual emission for aerosols approved for Esenshamm can be exceeded after a few hours by power house emission of Co 58 and without this emission being measured.

These results show by way of example that in addition to iodine isotopes, other nuclides as well must be taken into account in secondary loop emissions. This applies especially when primary loop activities multiply erratically based on the anticipated spikes.

Power house roof emissions not subject to measurement can also plausibly explain discrepancies between emission and immission measurements in the Obrigheim Nuclear Power Plant.

Between 1971 and 1975 radiation doses between 50 and 250 mrem/yr above the natural radiation level were measured in the vicinity of the Obrigheim Nuclear Power Plant using dosimeters by the State Institute for Environmental Protection of Baden-Wuerttemberg [12]. The unusual level of these values cannot be explained by the stack emissions of the Obrigheim Nuclear Power Plant during this period. As shown in Table 2.4.1.1, however, during this same period at the Obrigheim Nuclear Power Plant considerable steam generator heating tube leaks occurred to some extent which led to corresponding activity releases from the secondary loop above the power house roof. These emissions which were not reported and measured by the operator of the plant may explain the high radiation exposures in the vicinity of Obrigheim Nuclear Power Plant.

In summary it can be ascertained that steam generator leaks of 25 and 50 kg/hr under realistic conditions lead to a case in which allowable immission burdens are far exceeded by I 131 alone. Releases of iodine isotopes I 132, I 133 and I 135 exceed the emissions of I 131 and their immission burdens are added to the computed immission burdens by I 131.

The examples of tritium and Co 58 show that other nuclides than the iodine isotope must be taken into account in secondary loop emissions and the resulting immission burdens.

These results may also explain the inconsistencies between the emission measurements and immission measurements of the Obrigheim Nuclear Power Plant in a simple manner.

5.3. Bibliography

1. RWE - Biblis Plant Management, letter dated 2/25/80.

2. Schruefer, E., "Strahlung und Strahlungsmesstechnik in Kernkraftwerken" [Radiation and Radiation Measurement Engineering in Nuclear Power Plants], Berlin, 1974.

3. Hannover TSA, "Safety Evaluation of the Grohnde Nuclear Power Plant. Radiation Exposure by Emission of Radioactive Materials with Vent Air from the Crohnde Nuclear Power Plant at Special Receiving Points", November 1977.

4. Baden Industrial Monitoring Union, "Activity Release During Accidents in Nuclear Power Plants with LWR Via the Vent Air Pathway Within the Framework of a Guideline to § 28, Paragraph 3, of the Radiologic Protection Ordinance (Status 12/14/79).

5. Urban, H. and H. Biesold, "Evaluation of Radiation Exposure of the Population by the Discharge of Radioactive Materials with Vent Air in the Vicinity of the Site for the Neckarwestheim Community Nuclear Power Plant", GRS A-137, March 1978).

6. Bleck-Neuhaus, J., "Evaluation of Radiation Exposure of Thyroid by Radioactive Iodine 131" for the Administrative Court of Baden-Wuerttemberg, January 1981.

7. Bleck-Neuhaus, J., "Can an Equilibrium Model Describe the Radioecology of Iodine 131?" Joint Radiologic Protection Conference of the FS and the SFRP, Lausanne, 9/30-10/281.

8. Franke, B., U. Boikat and R. Ratka, "Transfer of Radioactive Materials from the Ground into Plants".

   Steinhilber-Schwab, B. and B. Franke, "Exposure Pathways and Selected Examples", Radio Ecology Symposium of the Association for Environmental Problems and the Federal Health Office, Stuttgart, Oct. 15-16, 1981.

9. Ordinance concerning protection from injury by ionizing radiation (Radiologic Protection Ordinance), October 13, 1976, Federal Register 1976, No. 125.

10. Commission of the European Communities: "Radioactive Effluents from Nuclear Power Stations and Nuclear Fuel Repressing Plants in the European Community", Discharge Data 1972-76, April 1978.

11. BMJ 1979, Federal Ministry of the Interior, "General Computational Principles for Radiation Exposure in the Case of Radioactive Effluents with Vent Air and in Surface Water", Joint Ministerial Bulletin, 1979.

12. Tuefel, D., "Radioactive Contaminations in the Vicinity of Nuclear Power Plants - Analysis of Official Environmental Monitoring", Association for Environmental Protection at the University of Heidelberg, 1976. Expanded edition 1979.

6. Emission Computations and Evaluation for Accidents with Steam Generator Heating Tube Ruptures

- Introduction

Steam generator heating tube ruptures should be considered LOCA under certain boundary conditions.

Depending on the number of steam generator heating tube ruptures, extremely small or small LOCA are involved. Contaminated primary coolant reaches the secondary loop through steam generator heating tube failure. Some of the coolant is released to the environment in certain accident sequences.

It was shown in Section 2.4 that steam generator heating tube failure is possible in the steam generators used in the FRG and must be incorporated in risk analyses.

The safety philosophy that the designs of materials used in the FRG precludes steam generator heating tube failure can no longer be maintained.

The accident in the Robert F. Ginna (U.S.A.) pressurized water reactor [1] at the beginning of February 1982 shows in conformity with our results, that steam generator heating tube failure should be analyzed not only in conjunction with feedwater line rupture or main steam line rupture.

Minutes after the steam generator heating tube rupture, secondary side relief valves in the Ginna reactor opened by means of which radiation was released into the environment.

In order to minimize these emissions, the primary-side relief valve of the pressure vessel was repeatedly opened and closed; after the fifth repetition the valve, however, failed and remained in the "partially open" position. Fortunately it was possible to reach a cold shutdown of the reactor 33 hrs after the beginning of the incident.

This accident sequence again highlights the "valve failure" weakness after Harrisburg.

Accident sequences with steam generator heating tube ruptures are analyzed both qualitatively and quantitatively in the following section using a single-fault concept [2, 3]. In this case it is shown that allowable immission limits are far exceeded according to the Radiologic Protection Ordinance in case of incidents.

In addition a review of accident studies of the Baden TSA [4] is presented.

6.1. Application of the Single Fault Concept to LOCA by Steam Generator Heating Tube Ruptures

It has been established by the Federal Minister of the Interior in the safety criteria for nuclear power plants [2] and principles for application of the single fault criterion [3] that a single fault should be assumed following coolant losses for residual heat removal.

The text reads:

Safety Criteria for Nuclear Power Plants, Section 4, Criterion 4.3:

"Residual Heat Removal Following Coolant Losses:

A reliable redundant system for emergency cooling (emergency cooling system) of the reactor core for coolant losses must be available and designed in such a manner that for the rupture magnitudes, rupture sites, operating states and transients in the reactor coolant system under consideration

1. The emergency cooling system can perform its safety task even during maintenance processes with simultaneous occurrence of single fault in the system.

2. The specified limits for fuel elements, core internals and for the containment are not exceeded.

3. The general chemical reactions are restricted to a safe level."

The utilization of the single-fault criterion is specified in interpretations of the safety criteria for nuclear power plants of the same importance and force as the safety criteria. Under item (1) it reads:

"With a single fault, a random additional fault in the safety systems not recognized before it is called upon and which does not occur as a result of a demand case in standard operation or in the case of accident is involved. A fault occurs when a system component[3] of the safety systems has not performed its function when called upon. A possible operating error resulting in a fault in the safety systems is equivalent to a single fault.

Reasons for the assumed fault generally need not be specified.

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[3]The concept "system component" comprises all components of the functional unit itself and supply, positioning and auxiliary equipment necessary (and in some cases even redundant) for proper safety functions."

If the steam generator heating tube failure is assumed as the cause of a loss of coolant, the failure of the secondary side relief valve (failure to close after opening) should be required as single fault, for example, in the case of residual heat removal.

There are good reasons (for example effects of two-phase flow) to assume valve failure as the result of steam generator heating tube ruptures.

This yields the following accident sequence:

So much primary coolant reaches the secondary loop due to steam generator heating tube failure and increases the pressure there that triggering of the relief valve occurs.

The relief valve of the defective steam generator fails in the "open" or "partially open" position (single fault).

Release of radioactive substances into the environment is caused thereby.

Accident Sequence:

-- Steam generator heating tube failure (initiating event)

-- Triggering of secondary side relief valve of the defective steam generator

-- Failure of secondary-side relief valve in "open" or "partially open" position (single fault)

results from application of the single fault criterion to residual heat removal during losses of coolant and actually occurred in Ginna U.S.A., as concerns the first two items.

A detailed description of the accident sequence as well as emission computations under different boundary conditions and their evaluation are cited in the following.

6.2. Description of Accident Sequence

Primary coolant enters the secondary loop due to failure of the 10 steam generator heating tubes. After approximately 10 sec reactor scram occurs by the signal "N16 > max" and after another 15 sec delayed turbine trip.

At this time, approximately 25 sec after initiation of the accident, the trigger pressure for the secondary-side relief valves is reached.

According to the single fault criterion a failure of the relief valve in the "open" or "partially open" position is assumed.

Three different scenarios with different boundary conditions are established for the continued accident sequence. I 131 was selected as the principal nuclide. The design activity was admitted due to the transients which occur with spiking factors between 25 and 100.

Determination of Boundary Conditions Which Apply to All Scenarios

-- Only the activity release during the first 30 min after initiation of the accident was computed. This assumption is not conservative since emissions must still be expected even after this time.

-- Approximately 1 min after initiation of the accident (depending on the scenario somewhat variable) the main coolant pumps are shut down by the "pressurizer water level < minimum" signal. Since the "primary pressure < minimum" signal is also present, loop feed begins through the safety injection pumps. In this case only two (of four) injected pumps were taken into account.

This is not implicitly a conservative assumption since main coolant leak into the secondary part of the defective steam generator increases when three or four high-pressure injection pumps are taken into account.

-- During the first 25 sec after the beginning of the accident, steam emissions from the high pressure section of 1 kg/sec and from feedwater degasification of 0.5 kg/sec were assumed.

-- In order to determine the activity release via the relief valves into the environment, only the amount of primary coolant discharged from the relief

valve of the defective steam generator was considered.

-- It was assumed that after approximately 30 sec no more feedwater is fed to the defective steam generators as make-up.

-- The first 20 sec after the beginning of an accident the average primary coolant leak in the secondary loop (in the following called the primary coolant leakage) is approximately 310 kg/sec.

The primary coolant leak drops to 280 kg/sec between 20 and 25 sec after the beginning of the accident. Subsequently the amounts of leakage vary depending on the scenario.

Boundary Conditions for Scenario 1

A spiking factor of 25 for I 131 is assumed.

0-25 sec after the beginning of the accident

- Average residual moisture in steam generator 30%.

25-300 sec after beginning of the accident

- Average residual moisture in steam generator 20%.
- Primary coolant leak drops from 280 kg/sec to 80 kg/sec.
- Steam emission from the defective relief valve ("partially open" position) of the steam generator with heating tube failure (in the following called steam emission) is on the average 120 kg/sec.

300-1800 sec after beginning of the accident

- Average residual moisture in steam generator 5%.
- Primary coolant leak drops, on an average measures 40 kg/sec.
- Steam emission drops and on an average measures 20 kg/sec.

Boundary Conditions for Scenario 2

A spiking factor of 50 for I 131 is assumed.

0-25 sec after beginning of the accident

- Average residual moisture in steam generator 30%.

25-100 sec after beginning of the accident

- Average residual moisture in steam generator 20%.
- Primary coolant leak drops from 280 kg/sec to 140 kg/sec.
- Average steam emission 150 kg/sec.

100-1800 sec after beginning of the accident

- Average residual moisture in steam generator 10%
- Primary coolant leak drops and on the average measures 70 kg/sec
- Steam emission drops and on the average measures 50 kg/sec.

Boundary Conditions for Scenario 3

A spiking factor of 100 for I 131 was assumed.

0-25 sec after beginning of the accident

- Average residual moisture in steam generator 30%.

25-1800 sec after beginning of the accident

- Average residual moisture in steam generator 20%.
- Primary coolant leak drops and on the average measures 140 kg/sec.
- Steam emission drops and on the average measures 150 kg/sec.

It would be desirable to examine in detail these as well as other scenarios by continuing studies, if necessary, using measurement programs.

In this case it would be especially important to ascertain at what number of steam generator heating tube ruptures (relief valve in "open" failure position) a core meltdown would occur. In this case some of the core inventory could be released directly into the environment by means of which the emissions of the aforementioned scenarios would be multiplied.

6.3. Computation of I 131 Emissions and Their Estimation

Activity releases were computed with the SEKEM4 program under the stipulated boundary conditions.

Figure 6.3.1 which follows shows emissions as well as emission rates during the individual time intervals for scenarior 2 by way of example.

The emissions relate to the time period described.

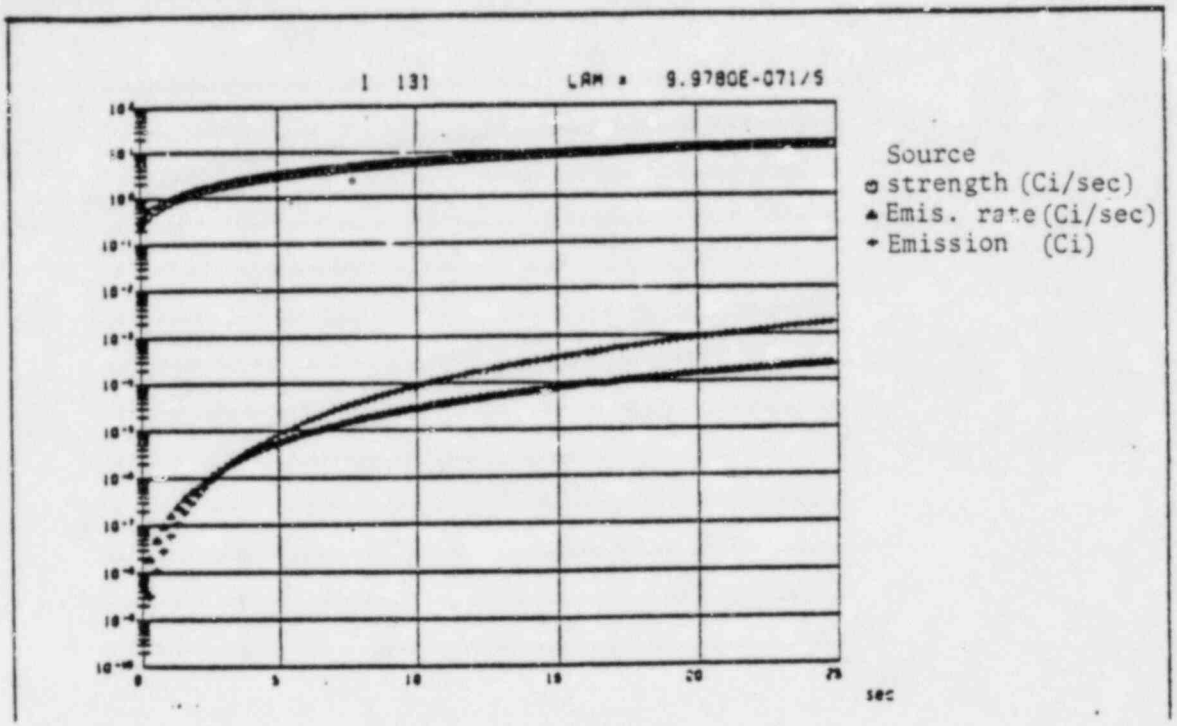


Figure 6.3.1. Emissions or Emission Rates of Accident According to Scenario 2. To the extent given, the source strength is the product of the time-dependent primary concentration and time-dependent leak rate.

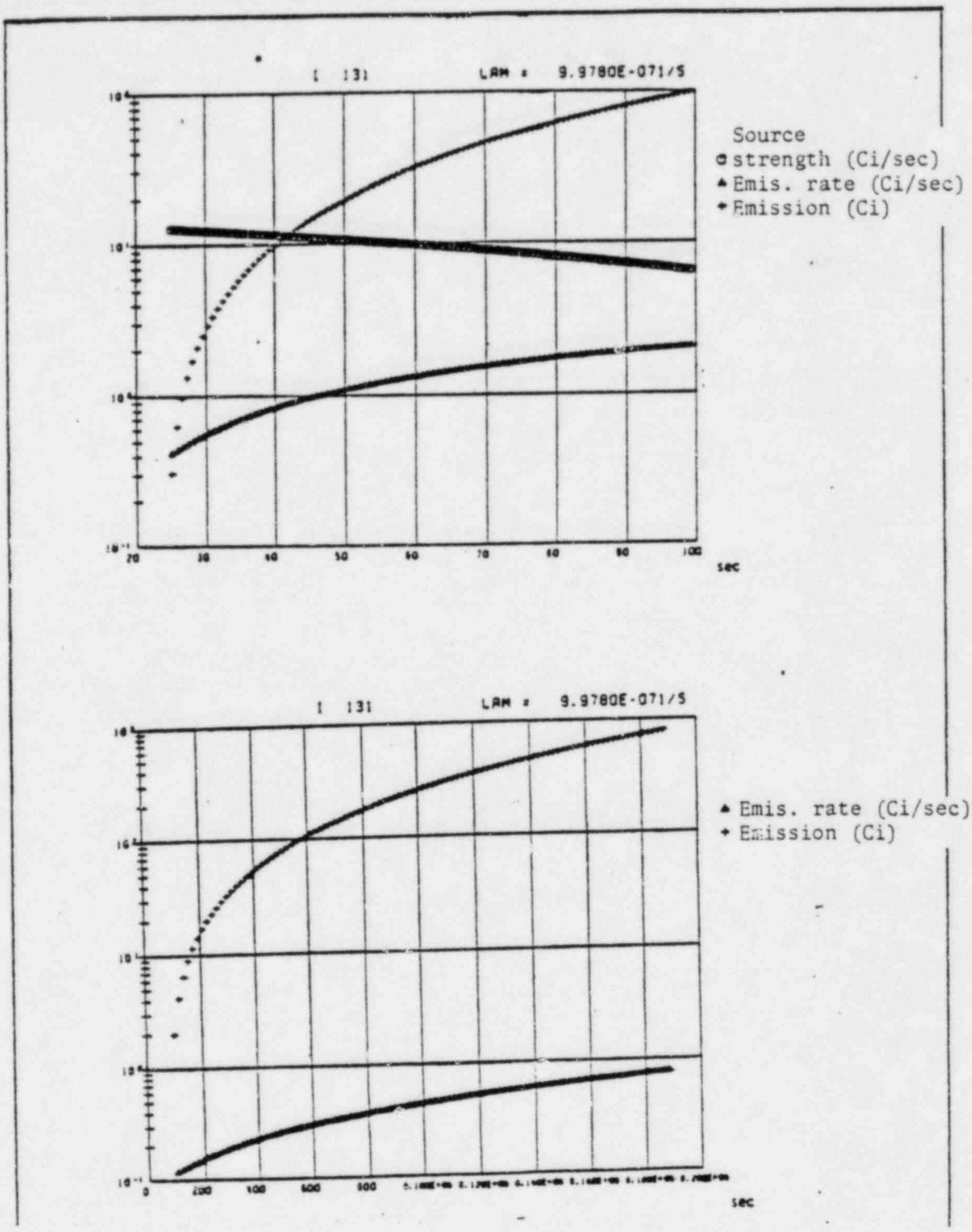


Figure 6.3.1. Emissions or Emission Rates of Accident According to Scenario 2. To the extent given, the source strength is the product of time-dependent primary concentration and time-dependent leak rate.

The table that follows lists I 131 emissions of individual scenarios during the first 30 min after beginning of an accident.

In addition the inhalation dose (only I 131) for adults as well as the thyroids of fetuses is stiluated under boundary conditions.

TABLE 6.3.1. EMISSION OF I-131 IN [CI] AND INHALATION DOSE [REM].

Accident scenarios	I-131 emissions in 30 min [Ci]	Inhalation dose for adults [rem]	Dose for fetal thyroid [rem]
Scenario 1	185	12	31 -- 250
Scenario 2	830	54	140 - 1100
Scenario 3	14400	940	2400 - 19800

Accident limit: 15 [rem].

Accident sequence: Steam generator heating tube failure and relief valve failure in "open" or "partially open" position.

The table shows that the accident limit of 15 rem for fetuses in all cases and for adults in scenarios 2 and 3 is exceeded by inhalation alone.

In the case of scenario 2 the limit for adults is exceeded by a factor of 3.6 and in scenario 3 by a factor of 60.

Exposure by all other emitted nuclides is added.

It was assumed that in the case of these accidents the consumption of food from the contaminated areas can be officially prohibited.

Thus only the inhalation dose of emitted I 131 was computed. Consideration of ingestion exposures would multiply the dose values.

Computation of Accident Dose

Inhalation dose D under accident conditions is computed as:

$$D = A \cdot \chi \cdot V \cdot D/Qi,$$

with A = amount of released radionuclides [Ci]

χ = short-term propagation factor [sec/m^3]

V = Respiration rate [m^3/Ci]

D/Qi = Inhalation factor [rem/Ci].

The following values were used:

-- The released I 131 activities are listed in Table 6.3.1 for the individual scenarios.

-- The Hannover TSA cites a value of 2 x 10^{-4} (sec/m^3) at a distance of 150 m as the short-term propagation factor for discharges from the power house. In this case the buoyancy of the cloud of activity is already taken into account by the simultaneously released large amount of superheated steam.

-- The value for adults, 20 m^3/d (≙ 2.31 · 10^{-4} m^3/sec) as the respiration rate was adopted from [7].

The inhalation dose factor for the thyroid (I 131), 1.4 · 10^6 rem/Ci was also derived from the same source.

As described in detail by Steinhilber-Schwab and Franke [8] the thyroid of the fetus represents the "least favorable influence site" via the "inhalation of I 131" exposure pathway.

Table 6.3.2 shows the ratio of fetal thyroid dose to reference value according to [7].

Accordingly the fetal thyroid dose exceeds the reference value by a factor of 2.6 to 21.

TABLE 6.3.2. COMPARISON OF I-131 INHALATION DOSE OF ADULTS, CHILDREN AND FETUSES (ACCORDING TO [8]).

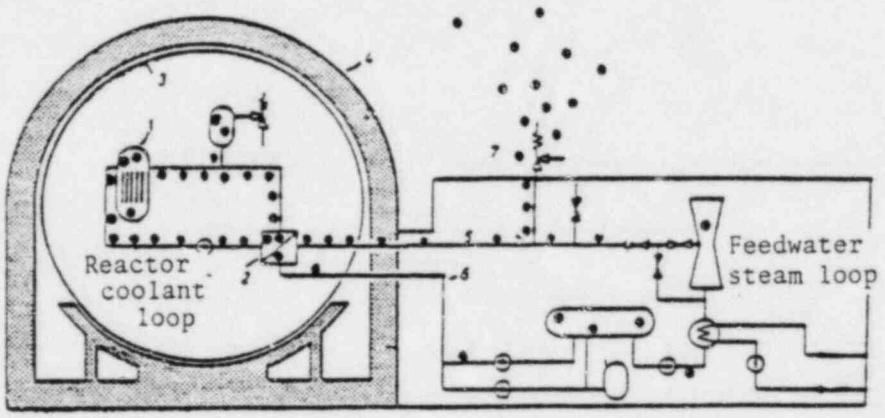
	Dose factor (BMI 1979) (in mrem/pCi)	Respiration rate	Dose for iodine 131 air concentration of 1 pCi/m^3 (in mrem/d)	Relative thyroid dose, adults (BMI) = 1
Adults	1.4 x 10^{-3}	20 m^3/d	2.8 x 10^{-2}	
Children	1.2 x 10^{-2}	2 m^3/d (in quiescent state ICRP 23)	2.4 x 10^{-2}	0.85
		6 m^3/d (in activity ICRP 23)	7.2 x 10^{-2}	2.6
Fetus	I 131 concentration in fetal thyroid 2-12 times that of the adults according to BMI (1979)	27 m^3/d (for slight activity ICRP 23)	7.5 x 10^{-2} to 4.5 x 10^{-1}	2.6 - 16
		36 m^3/d (for increased activity ICRP 23)	1.0 x 10^{-1} to 6 x 10^{-1}	3.5 - 21

In summary it can be ascertained that the accident immission limits can be exceeded several-fold when using the single-fault criterion by means of the LOCA under consideration. In the case of adults the limit is exceeded by a factor of 60 only by inhalation of I-131.

In the described accident sequence: steam generator-heating tube rupture (initiating event), failure of relief valve in "open" position (single fault), core melt-down accidents resulting in considerably greater releases of activity cannot be precluded.

This accident sequence should be assigned particular importance in future risk assessments. In addition the accident would have to be considered a design accident in the nuclear licensing procedure.

The dangerous aspect of this accident is the fact that the activity is released via steam generator-relief valve through almost all safety barriers into the environment (compare Figure 6.3.2).



1 Reactor pressure vessel
2 Steam generator
3 Containment
4 Reinforced concrete shell
5 Main steam pipes
6 Feedwater pipe
7 Relief valves (live steam safety valves and blowdown control valves)
0 Activity

Figure 6.3.2. Activity Release Under the Assumption All Safety Barriers are Bypassed. Accident sequence: Steam generator heating tube failure - relief valve in "open" position.

6.4. Review of Accident Studies of the Baden Technical Supervisory Association [TSA]

The Baden TSA analyzes accidents with secondary loop emissions (power house roof) in its study "Activity Release During Accidents in Nuclear Power Plants with LWR Via Vent Air Pathway Within the Framwork of a Guideline to Section 28, paragraph 3, of the Radiologic Protection Ordinance (status December 14, 1979)". Here the KKP II Plant currently under construction is used for analyses of the KWU pressurized water line (Philippsburg, Baden).

Two accidents of interest in conjunction with this study are discussed in detail from the list of accidents examined in the cited analysis. They are:

1. Rupture of the steam line outside the containment with simultaneous failure of 10 steam generator heating tubes.

2. Rupture in feedwater system outside the containment with failure of one steam generator heating tube.

There has been no opportunity to check the KWU programs modified by the Baden TSA and the systems analysis upon which they are based. Therefore the correctness of the computer programs as well as the computer plot with regard to the algorithm in addition to the systems analysis is assumed a priori.

Therefore accident sequences can be detailed only to the extent that special emission-relevant items occur therein. Data concerning activity releases receive special attention here.

6.4.1. Accident 1 - Rupture of a Main Steam Pipe Outside the Containment with Simultaneous Failure of 10 Steam Generator Heating Tubes

The TSA computes the amount of activity released above the power house for 30 min as a result of main steam pipe rupture with simultaneous steam generator heating tube failure (10 heating tubes) at the beginning of the accident from the main steam pipe leak and then via the atmospheric exhaust stations of the steam generator.

The accident is not considered conservative since only the failure of 10 steam generator heating tubes in one steam generator is assumed. There are more than 4,000 U-shaped heating tubes overall in a steam generator. According to our findings the failure of at least 20 heating tubes must be assumed for the transients which occur.

Emission is derived from the amount of primary coolant and its specific activity released into the environment.

In the TSA computation the primary coolant release of 191.63 kg with the specific activity of 21.6 Ci/t with an iodine 131 fraction of 0.92 Ci/t is assumed.

Both data are questionable.

1. The primary coolant discharge of approximately 200 kg is too low by a factor of approximately 27 as can be checked from in-house TSA data and assumptions. In this case correction of the untenable assumptions concerning steam generator residual moistures has not been undertaken.

2. The specific activity of the primary coolant of 21.6 Ci/t is stipulated by the TSA is too low at least by a factor 4.6 for a medium spike and 19.3 for a maximum spike.

This stipulated specific activity of the primary coolant can be exceeded in a short time as a result of load changes during normal operation (small transients). This spiking effect increases with the magnitude of the transients. Under the assumed accident conditions with large pressure transients, it can be expected that design activities will be maximally exceeded. Therefore design activities must take into account the spiking factors (compare Section 2.3.1.2).

Above and beyond the spike aspect, it must be considered that in TSA Tables 3.22 and 3.23 to which all computations of activity release relate not all relevant nuclides are included and the data consistently ignore tritium activities.

In conjunction with the accident discussed here, main steam pipe rupture with steam generator heating tube failure, let us suggest analysis of this accident with an additional relief valve failure in the "open" or "partially open" position in accordance with the single fault criterion.

In this case heating tube failure could be regarded as an induced fault of the main steam pipe rupture. Considerably higher emissions could be expected in this accident based on relief valve failure.

6.4.2. Accident 2 - Rupture in Feedwater System with Simultaneous Failure of a Steam Generator Heating Tube

Following rupture of the feedwater line, the contents of the feedwater system are drained at its lowest point together with the percentage of main steam through the evaporator overhead condenser to the power house roof (a total of 730 t) in which case some (approximately 170 t) is vaporized and discharged into the environment above the power house roof.

A total of 88 kg of water with primary loop activity reached the outfall ditch via the power house sump system. It is assumed by the TSA that these 88 kg of heating tube leak reached the feedwater tank with main steam via the evaporator overhead condenser up to delayed turbine trip (30 sec).

In this regard the following can be ascertained:

1. The accident sequence contains some non-conservative assumptions:

Only a single heating tube failure is assumed. Even in the case of rupture of only two heating tubes, the trigger pressure of the relief valves (86 bar for KKP II) would be exceeded. Considerable additional emission would occur (compare accident 1 - main steam pipe rupture).

As detailed in Section 2.4, without previous operating transients spontaneous failure of steam generator heating tubes can be expected. Therefore failure of at least 10 heating tubes can be assumed especially during accident-induced transients. In this case the conservative assumption would also have to be made that not all heating tube leaks occur in only one steam generator.

2. The computation of the activity release is contradictory.

2.1. Thus it is assumed that 88 kg primary coolant are discharged through the feedwater line leak without some, together with the 170 t, being vaporized and discharged above the roof.

If it is assumed that the 88 kg are uniformly distributed within the 730 t of total leakage water, 20.5 kg primary coolant would have to be vaporized concomitantly. In this case it is assumed that 170 t of the vaporized amount are correctly computed. The activity of at least 2.1 Ci sufficient for this amount of primary coolant for medium, and 8.5 Ci for maximum spikes, was not taken into account by the IMU.

2.2. The equation used to compute the main steam activity during operating leaks is inadequate. The equation does not take into account activity backflow from the feedwater tank into the steam generator (as Section 4, Sensitivity Analysis, shows, this backflow is considerable for the main steam concentration).

2.3. The main steam activities cited in Table 3.22 indicate discrepancies for those values which are computed from the given equation. A comparison of main steam values with the data of Table 3.23 "Nuclide Specific Activity in Steam Generator Water" indicates that the TSA has utilized excessively high decontamination factors throughout, measured at the current state of the art. This results in an underestimation of emission.

In summary it can be ascertained that the accident sequences of the two examples analyzed here are not conservative in any respects. This leads to underestimation of emissions. This applies especially to the number of failing steam generator heating tubes and primary loop activities. The accident boundary value can be exceeded by I-131 by considering the aforementioned factors.

6.5. Bibliography

1. NEWSWEEK, February 8, 1982, p. 43 ff, "Again a Nuclear Mishap".

2. Publication of Safety Criteria for Nuclear Power Plants, October 21, 1977, Federal Minister of the Interior - RSI6 - 513301-1/9.

3. "Interpretation of Safety Criteria for Nuclear Power Plants", publication of Federal Minister of the Interior - October 26, 1978 - RSI6 - 513301-4/1.

4. TSA Baden, "Activity Released During Accidents in Nuclear Power Plants with LWR Via Vent Air Pathway Within the Framework of the Guideline to § 23, Paragraph 3, of Radiologic Protection Ordinance", Status December 14, 1979.

5. "Effects of Serious Accidents in Reprocessing Installations and Nuclear Power Plants", reprint and interpretation of two confidential studies of the Institute for Reactor Safety, August and November, 1976, published by: BBU, 1977.

6. IMU Hannover, Safety Evaluation of Grohnde Nuclear Power Plant, SOG 1, "Radiation Exposure by Emission of Radioactive Materials with the Vent Air from the Grohnde Nuclear Power Plant at Special Receiving Points", Hannover, November 1977.

7. Federal Minister of the Interior, "General Computational Principles for Radiation Exposure in the Case of Radioactive Effluents with the Vent Air and in the Surface Water", Joint Ministerial Bulletin, 1979.

8. Steinhilber-Schwab, B. and B. Franke, IFEU, Institute for Energy and Environmental Research, Heidelberg, "Exposure Pathways and Selected Examples", paper presented at the Radioecology Symposium of the Association for Environmental Problems and the Federal Health Office, University of Stuttgart, October 15-16, 1981.

7. Conclusions

7.1. Consequences from Results of the Study

Emission computations for operating leaks in the steam generator as well as for steam generator heating tube ruptures show that the primary loop concentrations represent central input parameters.

In order to minimize risks associated with an increase of primary loop activities during normal operation, we propose that the reactor protection system be supplemented on the primary loop side by a continuously operating instrument to measure activity concentrations of individual major nuclides. When an established limit is exceeded, the reactor should be automatically shut down (compare Bibliography [5] and [1]). Osetek et al. in [1] proposes a special gamma-spectrometer with which fission product activity of the primary coolant can be continuously measured.

The accidents analyzed with steam generator heating tube ruptures show that the corresponding secondary emissions have been previously quite underestimated:

If we take into account additional failure of the secondary-side relief valve in the "open" or "partially open" position as we use the single-fault criterion for steam generator heating tube rupture, computations show that the accident limit is exceeded by inhalation for iodine 131 by a factor of 60 in adults.

These results call into question the statement of the "German Risk Study" [3]:

> "A leak in the pressurizer or in the steam generator itself is controlled by the safety system just as a leak in a main coolant line."

This statement does not apply with regard to the aforementioned accident which we propose in an application of the single-fault criterion to include as a design accident in the series of accidents taken into account within the framework of guideline to § 28, paragraph 3, of the Radiologic Protection Ordinance, since spontaneous steam generator heating tube failure cannot be precluded even without major transients (compare the Ginna accident, U.S.A., February 1982).

In the accident: steam generator heating tube rupture with failure of secondary-side relief valve in "open" position a core meltdown cannot be precluded as a result of loss of coolant. This would result in release of some of the core inventory via the secondary-side relief valve. The Institute for Reactor Safety (IRS) has studied the effects for the environment in detail [2].

In order to minimize the disastrous consequences of an accident of this type (the concept "accident" is intentionally selected since the accident sequence is dictated by the application of the single-fault criterion to LOCA), we propose construction of a containment in order to prevent direct release from the secondary-side relief valve into the environment. In this connection let us also mention construction of a subterranean safety tunnel into which the relief valves can discharge steam [4].

In addition, the following conclusions should be drawn from the results of the study:

-- The air conditioning plant in the valve building for the control room,etc. receives its humidity (approximately 160 kg/hr) from the auxiliary steam supply system. The auxiliary steam supply system itself is supplied either by the high pressure tap system, the main steam system or the auxiliary steam generator system depending on the operating state of the power plant. This should be modified immediately since in case of steam generator heating tube leaks the activity which reaches the main steam is distributed by the air conditioning system.

-- The scram limit of the N-16 signal should be coupled to the reactor power since otherwise at 50% power compared to 100% power steam generator leakage can be approximately twice as high before N-16 scram triggers.

-- Not only residual moisture transport but also direct steam transport should be taken into account especially in the steam generator for most nuclides since otherwise both the main steam activity and the emissions are underestimated.

-- Adherence to statutory maximum main steam concentrations for I-131 should be checked by continuous measurements. Otherwise excessive values can be expected.

-- Power house emissions would have to be measured by instruments. As long as this is not the case, repeated occurrence of differences between emission and immission measurements as in Obrigheim cannot be precluded since unmeasured activity releases can occur.

7.2. Bibliography to 7

1. Osetek, D. J., R. A. Coates, L. O. Johnson, E. W. Killian and J. J. King, "The Power Burst Facility Fission Product Detection System", NRC Instrumentation Review Group Meeting, July 1979.

2. "Tne Effects of Serious Accidents in WAA's and AKW's", reprint and interpretation of two confidential studies of the Institute of Reactor Safety, August and November, 1976, BBU, January 1977.

3. Reactor Safety Corporation, "German Risk Study of Nuclear Power Plants", Vol. 7, p. 36, published by the Rhineland IMU, 1980.

4. Akesson, B. A., S. G. A. Bergman and K. I. Sagefors, "WP Cave - A New Concept for Underground High Level Nuclear Waste Repository", Subsurface Space, Rockstore 80, Vol. 2, Pergamon Press, 1980.

5. Jakobi, S. and K. Tradowsky, "Cladding Monitoring on Fuel Elements of Nuclear Reactors", KfK 1452, October 1971.