SECONDARY LOOP RELEASES OF LIGHT WATER PRESSURIZED REACTORS IN NORMAL OPERATION AND IN SELECTED BREAKDOWNS

Providing of computer program SEKLMA - Sensitivity analysis of secondary loop releases - Release calculation in the case of failure of steam generator tubes and the assessments of results

IFEU - INSTITUTE OF ENERGY AND ENVIRONMENTAL RESEARCH

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1. INTRODUCTION

The present study aims at developing a computer program for the determination of radioactivity in the secondary loop and the radioactivity release in the turbine hall. Secondary loop releases have not been until now measured technologically and are the consequence of heat exchanger tube leakage in a steam generator. In Germany steam generator leakages have already occurred in 1971, in nuclear power plants in Obrigheim, and during the preparation of this study there were leakages in nuclear power plant in Biblis and Stade . In the Ginna reactor, in the United States, massive steam generator leakages occurred , so much so that the secondary side relief valves had to be opened and radioactivity was released into the environment.

The program is tested with the aid of sensitivity analysis and the influence of the following parameters compared with the release rates :

primary loop activity, steam generator - heating tube leakage , radioactive decay constant, steam release rates from the high pressure side and from the feed-water degassing, residue humidification, (for instance, in the steam generator), elutration rates per steam generator, decontamination factors in the elutration demineralization plant as well as phase distribution factors in high and low pressure areas.

In addition, load changes and radioactivity spikes will be investigated. During the sensitivity analysis the influence of various parameters on release pathways, so far not recognized, will be investigated.

It is planned to carry out realistic computation of the plant leakage in a steam generator. In this matter special attention will be paid to what degree the officially fixed maximum J-131 fresh steam concentration shall be adhered to , and also what going above these values would mean. In addition, the releases should be assessed by emission calculations. In particular, we are going to find out whether other nuclides, in addition to iodine-131 are relevant in secondary loop releases.

The functioning capacity of the N-16-signal, especially the connection between the response time and the reactor power will also be investigated.

The single fault criterion will be applied to a coolant-loss disturbance caused by steam generator tube failure. It will be investigated whether the perturbations which result from tube failure were considered in the accidents discassed, and what were the consequences of them. With the aid of the computer program the release of these breakdowns will be calculated.

Introduction

Voluminous documentation studies show that secondary systems of the German nuclear power plants with pressure water reactors are comparable in their essential component parts. See (1), p.61, diagram 3 2.8., principal circuit diagram of the main heat removal system; (2) diagram 8, outline; (3), diagram 6, water steam loop; (4) especially page 12; (5) especially p.32 diagram 1, also 3,3 or 4; loop installations in principle are built in a similar way, so that in release calculations for the turbine hall in general only vary with mass flows and the varying efficiency of the elutration demineralization plants are taken into account.

See (6),(7),(8),(9) and (10).

In the investigation of the individual components, the research was based on the standard work by Karl Schröder (11).

In the scenario or models development, direct contact with the managers in addition to the study of the literature was essential, and also the examination of internal documents of the nuclear power plant at Neckarwestheim as well as on-the-spot inspection and discussions at the nuclear power plant in Grafenrheinfeld, at Biblis A and B.

As a reference was chosen the npp Biblis Block B (See (12) p.35) although in earlier power plants some components are not correspondingly designed. Biblis B, however, has the advantage of corresponding to the plants in construction or planned (construction line 8) so that the computer program developed by us is applicable to them.

In the following pages we deal in detail with the scenario used in the study of the secondary loop of the light water reactors. In it we discuss the system-oriented as well as the operational technical marginal conditions. Then follows a discussion about the nuclide transport in SK determined magnitudes - which are the introductory parameters important for the computer program.

For the nuclide distribution a mathematical formula is deduced for local and nuclide specific distribution factors. At the end of this chapter a section will be devoted for the difficulties in information obtainement.

2.1. Scenario of the secondary loop of light water pressurized reactors

1. On the one hand, it must reproduce the main structure and function of the secondary loop in a simplified and summarized way, without falsifying or omitting the essential characteristics of the system in its functional context.
2. On the other hand, The scenario must make the mathematical formulation of the functional system possible.

The secondary loop scenario is also the result of a system analysis. It must contain the essential transport pathways for nuclides and especially the release pathways.

2.1.1 Structure and function of the pressurized water reactors

The description of the secondary loops - also called feedwater steam loop - is placed at the outset for the completion of the structural explanation and the way of operation of the nuclear power plant, with pressure water reactor.

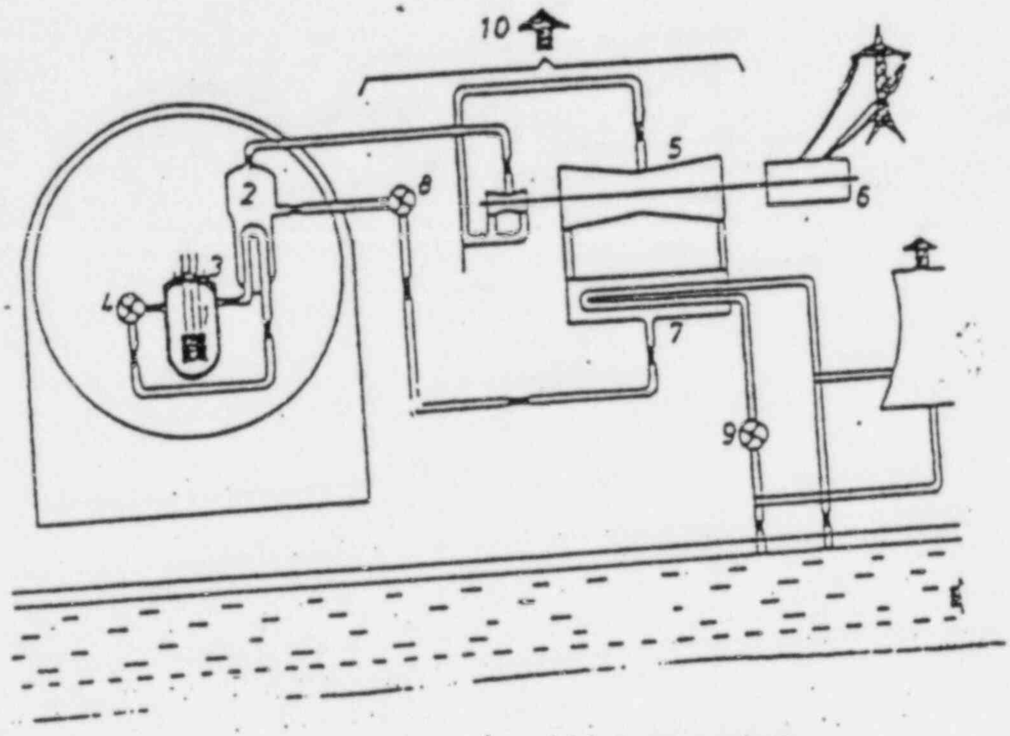
The structure and method of operation of a nuclear power plant with pressure water reactor is illustrated by the simplified operation diagram 2.1.1 . We see there three loops :

1. Primary loop : which is an essentially closed heat-transfer circuit in which the heat produced in the reactor co: 1 is transferred by the main coolant pump 4 from the reactor pressure vessel 3 via the steam generator 2 (At most 2 to 4 steam generators per reactor block) to the secondary loop.

2. Secondary loop : the feed water heat-transfer circuit largely corresponds to the conventional power plants. The feed water is pumped through it h by the main feed water pump into the steam generator, where it is evaporated by the heat from the reactor loop. The generated steam drives the turbine installation consisting of a high pressure turbine and three para low pressure turbines, and these in turn drive gener 6. The steam exiting from the turbines is condensed in condenser 7 by release of its heat to a coolant medium,ad and the loop is completed. Arrow 10 show that during the operation steam leakages continuous: occur through the turbine wall roof.

3. Tertiary loop : The main heat-transfer circuit is used to transfe heat from the condenser. During this operation, water is pumped by the main coolant pump through t condenser which releases about 2/3 of the heat proc in the reactor core, to be discharged, as the case be, directly into a river, into the sea or through the cooling towers into the atmosphere.

Chart 2.1.1 Symbolic Circuit of a Nuclear Plant with Pressure Water Reactor

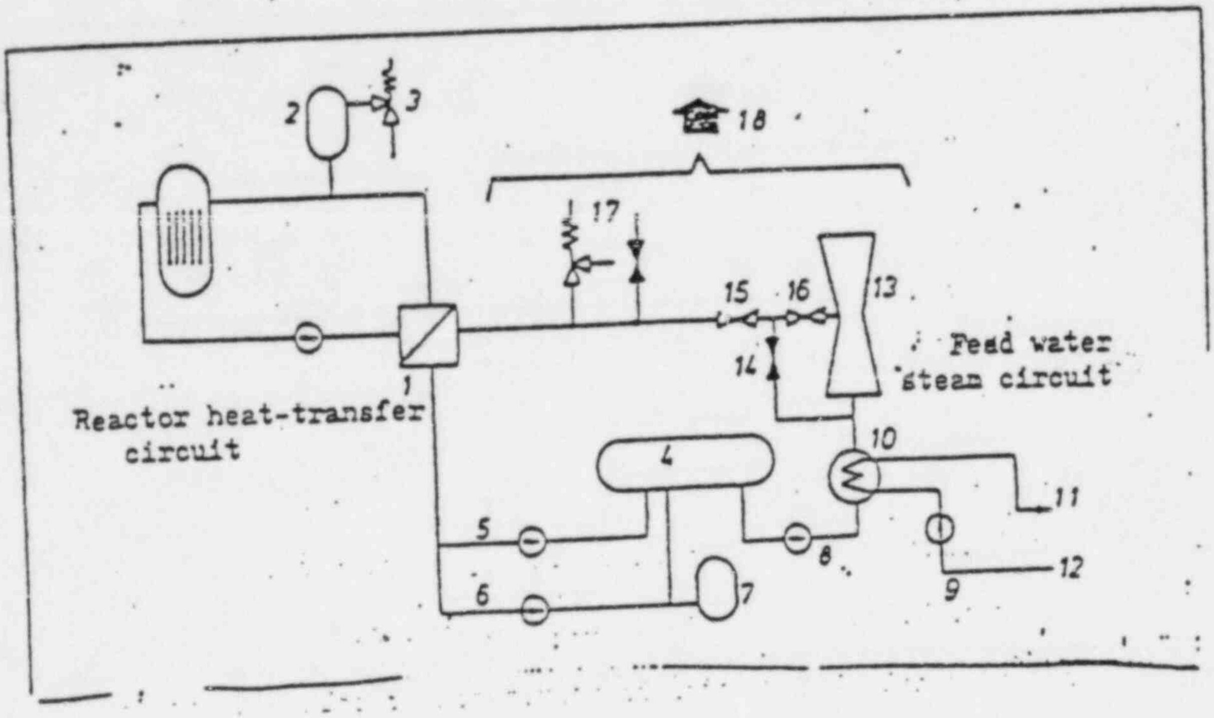


1. Reactor core
2 Steam generator
3 Reactor pressure tank
4 . Primary coolant pump
5 Turbine
6 Generator
7 Condenser
8 Main feed water pump
9 Main cooling water pump
10 Steam releases from the turbine hall

The generation of electric power from heat in nuclear power plants occurrs in principle in the same way as in other thermal power plants. The difference consists in the heat production through the process of nuclear fission. Since during this operation several hundred radio-active isotopes are produced which because of their radiation could cause poss injuries, and constitute a risk for living organisms, special precautions must be taken to minimize the release of radio-active materials into the power plant environment. Radioactive releases cannot be completely eliminated even during normal undisturbed operation.

2.1.2 Structure and Function of the Secondary Loop

Structure and function of the secondary loop are represented in a simplified form in the basic circuit diagram No.2.1.2



1 Steam generator	10 Condensor
2 Pressurizer	11 Discharge for condensor cooling water
3 Pressurizer emergency relief valve and blow-off valve	12 Inlet for condenser cooling water
4 Feed water container	13 Turbine
5 Main feed water	14 Steam bypass equipment
6 Emergency feed water system	15 Main steam valve
7 Hydrogen recombination tank	16 Turbine safety regulator valve
8 Main condensate flow	17 Steam safety valve and blow-off valve
9 Condensor cooling system	18 Steam emissions from the turbine hall

Chart 2.1.2 : Basic circuit diagram of the reactor heat-transfer circuit and the feed water steam circuit

The function of the steam and feed water systems is the transfer of nuclear heat and its thermokinetic transformation into electricity. The important operational elements are the steam generators, turbine and electric generators and the steam condenser.

As we have already sketched in brief, the fresh steam generated in the steam generators leaves the safety tank through the fresh steam piping provided with relief and safety valves, and flows through the turbine valves to drive the turbine unit. The steam from this operation is condensed in the condensers, the condensate flow is collected in the condensate tank and returned by the main condensate pumps through some prewarming steps into the feed water tank, and from there is pumped by the main feed water pumps back into the steam generator. During this time the steam for the prewarming of the condensate is tapped at various turbine stages in the way convent to all thermal power plants.

The complete technological representation of the secondary loop fc within the circuit diagrams :

1. reactor heat-transfer circuit
2. main feed water system
3. emergency feed water system
4. hydrogen recombination system
5. steam piping system.

Each of these circuit diagrams comprises about 6 - to 8 DIN 4 pag (See German Reactor Safety Study, Bibliography No.15) with several hundred components.

It is not meaningful for determination of releases from the second loop to go into great detail about its parts or their technologi aspects. As the following section shows, it is possible to keep the number of components sufficiently small for comprehension (treatment in a scenario.

2.1.3 Development of the working model

The permanent study serves the following general purpose : the computation on the basis of the primary loop radioactivity of any radio nuclide as well as of the steam generator leakage, the specific radioactivity of this nuclide, the radioactivity flow between two connected sites as well as the releases resulting from them from the secondary loop at any time and at any place of the secondary loop.

The following restricted function is relevant to the model development within the system analysis :

On the basis of the primary loop activity of any radio nuclide as well as of the steam generator in a state of equilibrium and for a determined operating transient, the computation has to be undertaken of the specific radioaativity of this nuclide for relevant sites and for releases from the secondary loop.

The task of model development consists in presenting a model which can exactly simulate continuous space/time distribution of any radio nuclide in the secondary loop. With regard to time distribution, steady state activities and in special cases time-dependent transient conditions must be taken into consideration .

With respect to space distribution an appropriate selection of sites in the feed water - steam system must be made.

Further, it must be determined which magnitudes for the nuclide transport are essential in the secondary loop.

Within the framework of model development the following nuclide specific and non nuclide specific determination magnitudes are taken at the start :

1. Specific primary loop radioactivity and steam generator leakage

The specific primary loop radioactivity is specific to nuclides, the steam generator leakages are not. These two magnitudes go into the radioactivity computation in a linear way. The present marginal and intitial conditions, which affect the height, but not the space and time distribution of the specific nuclide activity.

2. Mass flows and steam emissions

Mass flows and steam releases are not specific to nuclides. The radio nuclides carried through the steam generator leakage into the secondary loop are carried with water or steam; they are distributed according to the mass flows of water and steam phases among other things. This is expressed by the fact, that for each radio nuclide, specific activity, thus the number of decays per time unit and mass unit (most inCi/t) is computed in each as relevant considered site of the SK.

3. Phase distribution factor alpha or decontamination factor DF

Factors alpha and Df are specfic in nuclides and indicate in what quantity a radio nuclide remains in the transition phase of evaporation or condensation in a fluid or in gaseous phase. Thus, for instance, a nuclide which is ionized in an evaporation process will mostly remain in fluid phase. Factors alpha and DF depend on the chemical and physical properties of the considered nuclide, as well as on technically indicated magnitudes of pressure, temperature, steam humidity, pH-values , etc.

A detailed description of the magnitudes determining nuclide transport will be found in sections 2.2 a or 2.5 of the system analysis.

Since the model development is closely connected with mathematical formulation, mathematical description or solution statements let us consider the computer program here before chapter 3.

The following basic physical assumptions determine the mathematical treatment :

1. The mass conservation law or continuity equation applies for the mass flow or mass current density.
2. In the Sk-System (model) there are no other sources for radio nuclides outside the steam generator leakages, given by the leakage rate in the steam generator and in the primary loop concentration of cific nuclides.
3. For each site in SK the radioactivity can be estimated from the technically given mass flows by linear differential equation. The radioactivity distribution for all parts flows as the solution of differential equation systems [DE-Systems].

The number of places to be taken into account determines directly the extent of the DE-System and the expenditure for its solving.

The following places have been selected as indicated above and will be explained shortly (See-diagram 2.1.3 and diagram 2.1.4).

1. DE - leaking
2. De - leakproof
3. FD fresh steam
4. High pressure turbine
5. Water separator or trap/ intermediate superheater
6. Low pressure turbine
7. Condenser
8. Hot well
9. Feed water tank
10. The blow-down flash tank (elutration strees-reliever)
11. Elutration demineralization (elutrat on demineralization)
12. Sump
13. Atmosphere

With regard to 1 and 2 :

2

the steam generators play a central role from two points of view : on the one hand, the steam generators are the site in which radio activity is introduced from the primary loop into the SK. On the other hand, according to the element and the chemical form in which it is found, the increase in the activity in the fluid phase occurs in the evaporation process. Here the pressure , temperature and the residual humidity are of decisive importance.

This phenomenon, which rests on high differences in solubility of substances in water or steam, will be described in terms of the aforementioned phase distribution and decontamination factors. (See section 2.2 and 2.5).

We proceed from the assumption that a steam generator leakage does not occur simultaneously in all steam generators. Thus, in practice so far there have been one, at the most two, steam generator leakages among the four steam generators to be found in the reactor. The seenario must assume exactly two sites of steam generators, of which one is intact and the other represents a steam generator with heat exchanges tube leakages.

With respect to 3 : the fresh steam generated in the steam generator is dried and distributed in an extensive piping system. The practice of introducing fresh steam into a stage is based on the consideration that the fresh steam tubes of a steam generator connect into a gath ring line before its entry into the turbine which will influence the radioactivity distribut on. Further, some of the already collected fresh steam will be diverted and conveyed for the purposes of pre-heating into the 5th stage, bypassing the turbine, into the water separator and into the intermediate superheater, and from there into the feed water tank. This is important because of its feed water degassing device for the release computation.

With respect to 4 :

The high pressure turbine should be seen as a relevant site because , on the one hand, it constitutes a mass flow distribution junction with direct inflow into the feed water tank thanks to its numerous taps which we have indicated, and, on the other hand, it changes the magnitudes of pressures, temperature, steam quality and residual humidity relevant for the phase distribution factors.

With respect to 5 :

As with the high pressure turbine, so also the water separator/and intermediate superheater represent mass flow branching, which influences radioactivity flows over factor alpha with direct inflow into the SW tank. Pressure, temperature and humidity contents of the fresh steam are also influenced by these components.

With regard to 6 : see 4

With regard to 7 :

the condenser is an operational and safety technical power plant component because it fulfills a relevant function of heat removal, for instance, during a reactor "scram" or a turbine trip-out and operates through the bypass station. Further, it constitutes a linkage of the tertiary loop with the main water loop.

Condenser leakages decisively influence tubes of the steam generator. Finally, an essential release pathway for noble gases exits from SK through condenser absopption.

With respect to 8 :

the hot well represents the central point of the mass flow. Here the mass flows collect from the 1 elutration (blow-down) demineralization, low pressure turbine taps and condensate, and flow from the hot well into the feed water tank. In the hot well, the hydrogen recombination is in addition resupplied as a make up for the water and steam losses.

With respect to 9 :

A direct release pathway goes through the feed water degassing into the atmosphere . The feed water tank in addition is a mass flow junction point, with a direct line to the steam generators.

With respect to 10 :

The blow-down flash tank serves the pressure and temperature change of the steam generator water to the values required for the blow down flash tank installation. During the depressurization the water in the steam generator is generated, which is directly conveyed into the feed water tank. This mass flow is decisive in the radioactivity of the feed water tank and for its release.

With respect to 11 :

The elutration demineralization serves to hold-back nuclides in a crystalized state.

With respect to 12 and 13 :

The sump and atmosphere represent receiving phases for the water and steam.

The modell optimization described above takes into account in brief the following marginal conditions :

- the degree of specification in the description of the reactor technique
- determinant magnitude of the radio nuclide transportation
- mathematical formulation statement
- problem adopted EDV optimization (location, computation, time, etc.)

2.1.4 . Computation of Radio Nuclide Emissions for the Secondary Loop

The result of the scenario development is shown in diagram 2.1.3 As the mass flows are shown, the component relevant for the determining of mass flows and the wiring system are included. The scenario of the secondary loop for the computer program is derived from diagram 2.1.3 and shown in diagram 2.1.4.

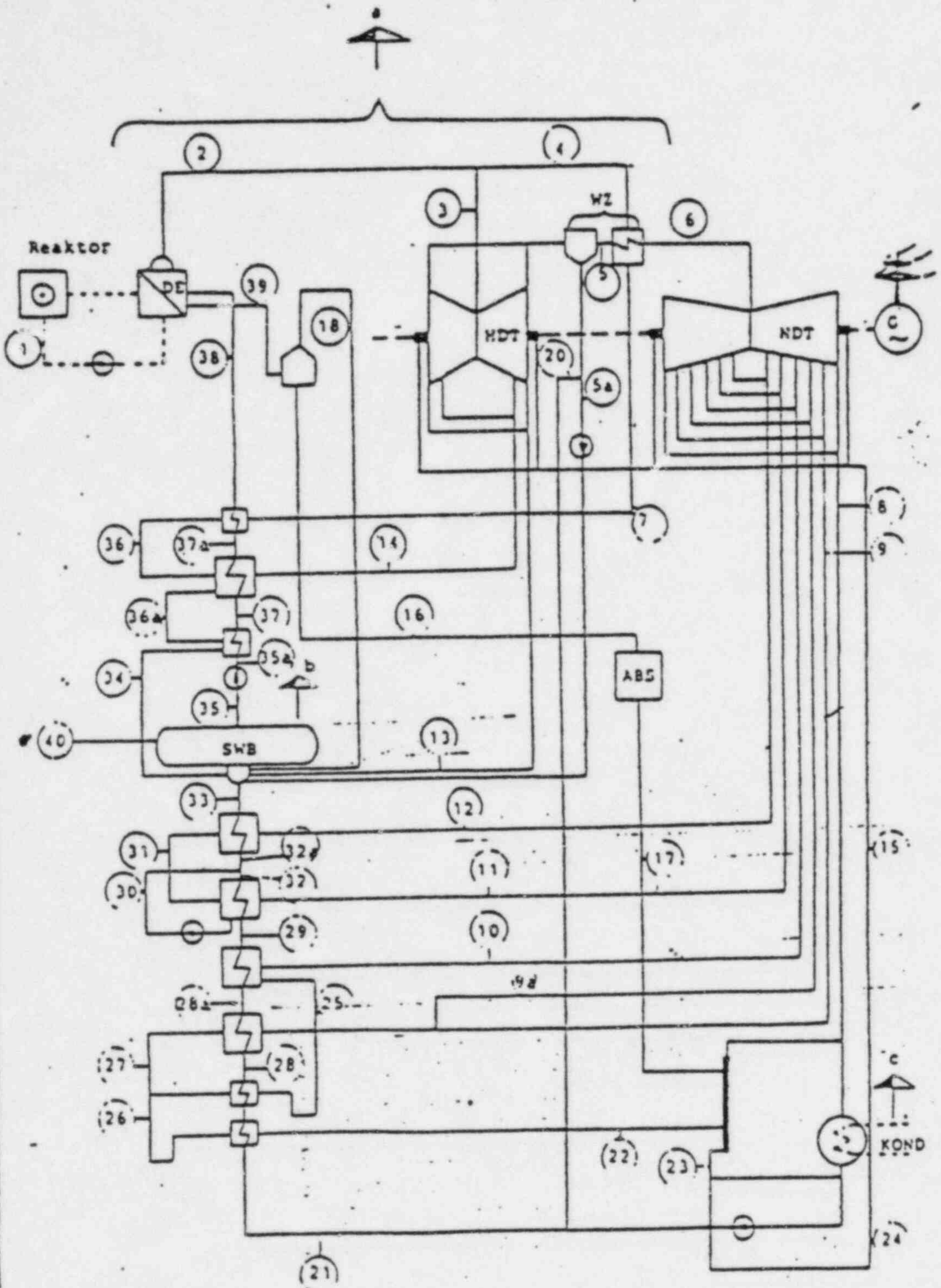
It becomes evident when we compare diagram 2.1.3 and 4 that the regulation and safety technical components such as tension valves, shut-off valves, safety accessories are omitted or summarized. Parallel piping in diagram 2.1.4 is either omitted - because it is of no importance for nuclide transport - or summarized.

Three release pathways have a special importance for the releases into the atmosphere , two of them take place not through the ventilation chimney :

1. The main leakage point at high pressure points, from which even in normal operation, emissions are released are indicated by the following brackets I—⌒—I Indifidually, the emission points are to be understood to include the following components : high pressure sump hole release, with its standing vent into the turbine hall roof and leakages from various waterproofings and glands.

2. Emissions from the feed water degassing.For reasons of water chemistry the feed water must be freed as much as possible from dissolved corrosive gases such as oxygen and carbon dioxide. The gases are released through the turbine hall roof with water steam as the carrier gas (without smoke cooler/ waste vapor condenser)

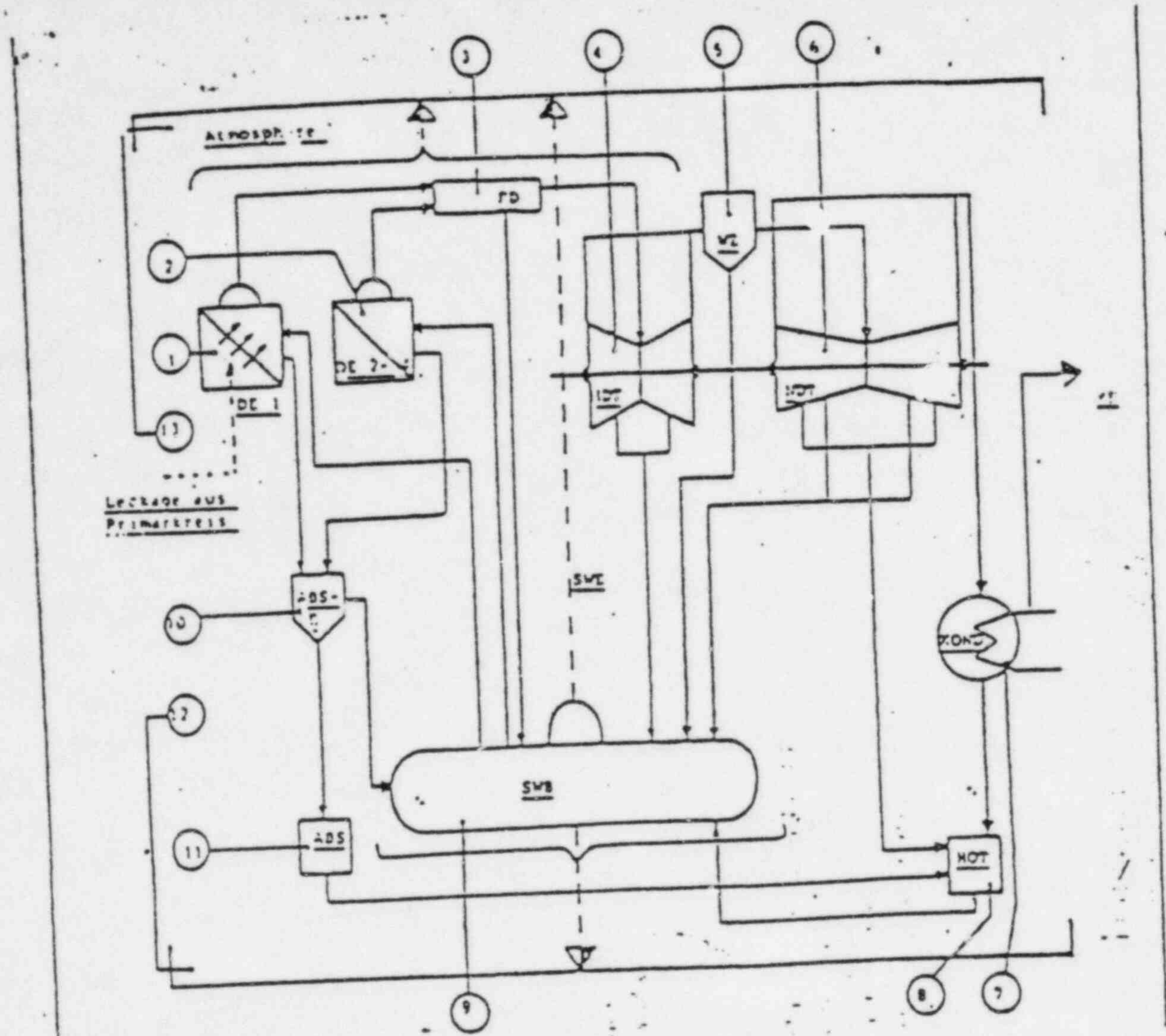
In this case it is important that about only half of the mass flows which are diverted for the purpose of demineralization from steam generators . reaches the demineralization installation.



a - releases in high pressure area unfiltered
b - releases during the feed water degassing

c - releases during condenser evacuation through chimney

IFEU: Institut für Energie- und Umweltforschung Heidelberg e.V. Im Sand 5 D-6900 HEIDELBERG	Diagram 2.1.3. Simplified model for for the computation of secondary loop releases in steam generator leakages

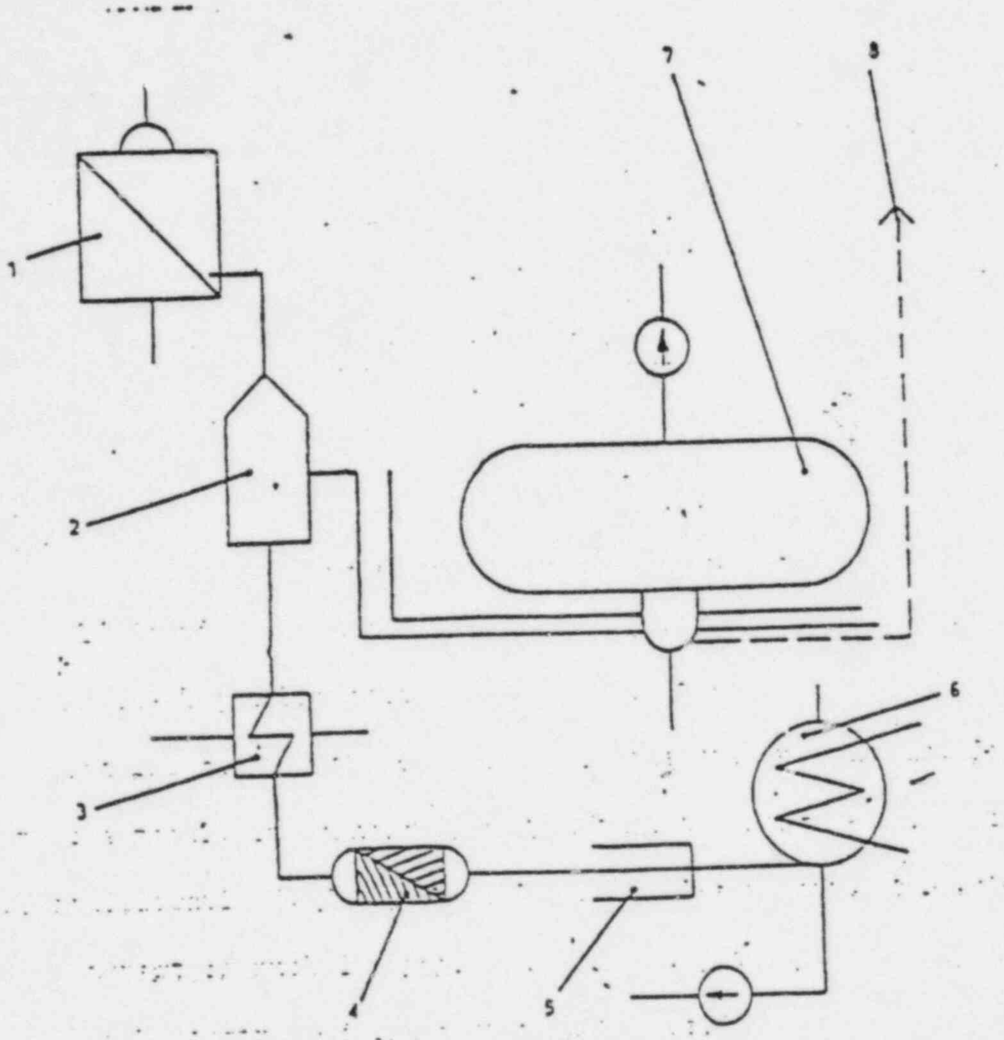


Places in computer program

1 DE - leaking
2 De - leakproof
3 FD - fresh steam
4 High pressure turbine
5 Water separator or trap/
intermediate superheater
6 Low pressure turbine
7 Condenser
8 Hot well
9 Feed water tank
10 The blow-down flash tank
(elutration stress-reliever
11 Elutration demineralization
12 Sump
13 Atmosphere

SWE - Feed water degassing
KE - condenser evacuation

Diagram 2.1.4 : Model of the Secondary loop
for the Computer Program



1 Steam generator
2 Stress relief of the elutration demineralization
3 Cooler of the elutration demineralization
4 Cleaning system of the elutration demineralization
5 Hot well
6 Condenser
7 Feed water tank
8 Release pathway : feed water degassing into the atmosphere

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Diagram : Elutration demineralization
2.1.5. in a steam generator

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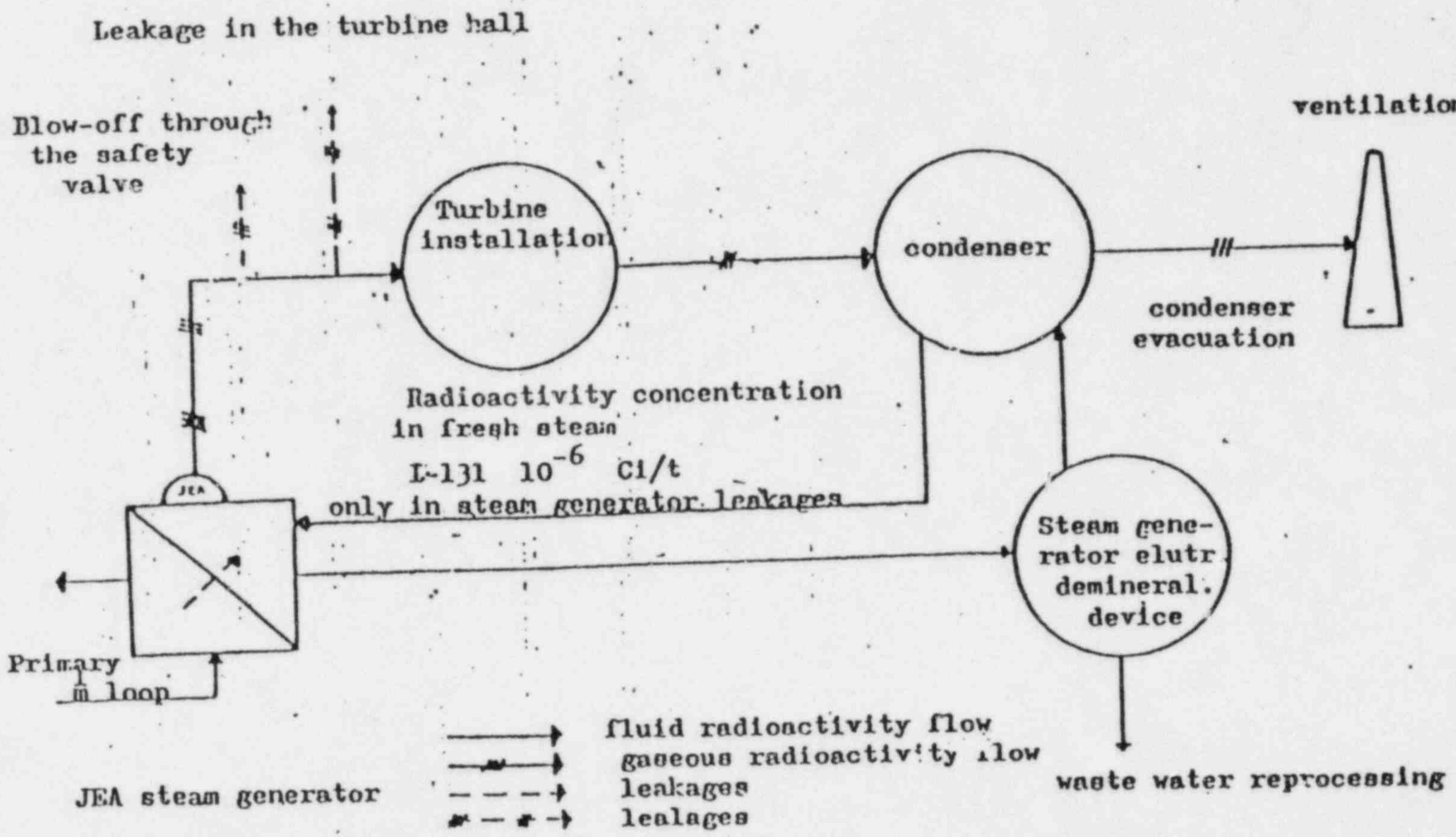
Diagram 2.1.6

Explanation diagram for radioactivities in the systems according to the bibliographical entry :



The other half of this mass flow reaches the feed tank unfiltered. In cases of steam generator leakage, radioactivity goest directly into the environment through this pathway via the feed water degassing (See diagram 2.1.5.).

3. Releases from the condenser absorption:

Noble gases under pressure are released through the hot vent.

2.1.5. Theoretical Comparison of the Sk Model with the Secondary Loop Model of the Union of Nuclear Power Plants (KWU)

At the completion of the system analysis we must briefly examine the SK Model use by the Union of the Nuclear Power Plants (KWU).[13]. The theoretical comparison is informative on the one hand, because the KWU-Model found acceptance in the evaluation of nuclear power plants within the legal licencing procedure, on the other hand, because the comparison illuminates the importance of the system analysis. It is evident that a model, in principle, cannot take the release pathways more into consideration * of all the relevant system magnitudes - in this case the transport of nuclides including release pathways the derived model cannot completely comprehend fragmentary releases.

The KWU diagram for the computation of radioactive flows in the secondary loop is represented in diagram 2.1.6.

A detailed comparison reveals :

1. The KWU - Model has no feed ~~wt~~ water tank, but rather a condenser, and hence no feed water degassing. As a consequence all of the live steam goes into the condenser through turbines which must lead to the overestimation of those releases, which exit thr ugh the roof stack and thus in a "controlled" way.

* than can be included within the framework of the system analyses. This means, that without a complete consideration

Subcommittee on Environment,
Energy and Natural Resources

Translation (German)

6. COMPUTATIONS OF RELEASES AND EVALUTTION IN BREAKDOWNS CAUSED BY BURST TUBES IN STEAM GENERATORS

Introduction

The bursting of tubes in steam generators must under certain marginal conditions be evaluated as a breakdown caused by the loss of coolant.

According to the number of tube breakdowns in steam generators, we may differentiate between minimal and small breakdowns caused by the loss of coolants. As a result of tube breakdowns in steam generators, the secondary loop is infiltrated by primary coolants of which part is released into the environment in certain breakdown sequences.

In Section 2.4 it was pointed out that tube breakdowns in steam generators may also occur in steam generators used in the Federal Republic of Germany and must be taken into consideration in evaluations of.

The viewpoint that tube breakdowns in steam generators will not occur ~~constructions and materials used in the Federal Republic of Germany~~ can no longer be maintained.

The leak at the Robert F. Ginna (U.S.) pressurized water reactor (1) at the beginning of February 1982 shows, in agreement with our results, that the breakdown of tubes in steam generators must not only be viewed in connection with the bursting of the feed water line and the live-steam line. In the case of the Ginna reactor, secondary discharge valves opened a few minutes after the bursting of the tubes in the steam generator as a result of which radioactive particles were released into the environment.

In order to minimize such releases, the primary discharge valve was opened a few times and closed again - after it was opened for the fifth time, the valve failed and remained in a "partially open" position. Luckily, 33 hours after the breakdown the reactor was brought under control by a cold shutdown.

225 After Harrisburg, this breakdown points again to the weak point caused by "valve failure."

In the following chapter breakdown sequences caused by ruptured tubes in steam generators will be discussed qualitatively and quantitatively as well as the application of the concepts of individual error (2,3). It shows that, in accordance with the regulations on radiation protection, the admissible ~~emission~~ emission cut-off values have considerably been exceeded in breakdo

In addition, there ~~follows~~ is a critical discussion of the studies of breakdo carried out by the Baden Technical Control Board (4).

6.1 Application of the Individual Error Concept in ~~Accidents~~ Coolan Loss Accidents Caused by Burst Tubes in Steam Generators

In the safety regulations for ~~atomic~~ nuclear plants (2) and ~~the~~ principles f the application of the individual error criterion (3), the Federal Ministe the Interior has established that an individual error must be assumed in post heat discharges aft loss of coolants.

The exact wording is:

Safety Criteria for ~~Atomic~~ Nuclear Plants,

Section 4, Criterion 4.3:

" Post heat discharge after coolant losses:

A reliable and redundant system for the emergency cooling (emergenc cooling system) of the reactor core in coolant losses must be available and operate in such a manner that, with regard to sizes, locations of rupture points, working conditions and transients in the reacto cooling system

1. the emergency cooling system will safely perform ~~[illegible]~~ also during maintenance work and ~~[illegible]~~ simultaneous occurrence ~~[illegible]~~ of an individual error in the system.

2. the specified cut-off values ~~[illegible]~~ for ~~[illegible]~~ fuel elements, installed core ~~[illegible]~~ equipment and safety be ~~[illegible]~~ not exceeded.

3. the chemical reactions be limited to a degree ~~[illegible]~~ absolutely safe from the technical point of view.

226 In the interpretations of ~~[illegible]~~ the safety criteria for nuclear ~~[illegible]~~ plants which have the same significance and binding character as the safety criteria themselves, the application of the individual error criterion is specified as follows under Point (1) which states verbatim:

"The individual error is considered to be an accidental error which must not be equated with an error that occurred as a result of the operational requirements or of any additional error ~~[illegible]~~ in the safety installations. ~~[illegible]~~ An error exists if part of the system [3] of the safety installations does not ~~[illegible]~~ functions properly during operation. A possible ~~[illegible]~~ ~~[illegible]~~ false operation resulting in an ~~[illegible]~~ error in the safety installations must be equated with an individual error.

Generally reasons for an assumed error must not be given."

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3) The term "part of the system" covers all the parts of the operational ~~if necessary redundant~~ unit itself and the supply, adjustment and and if necessary, redundant auxiliary installations ~~that are necessary~~ for its safe operation.

If the breakdown of a tube in a steam generator is considered to be the cause of a coolant loss, we must regard as an individual error, for instance, the failure of a secondary discharge valve (failure to close after it has been opened) in post heat discharges.

There are good reasons (for instance, the resulting effect of a two-phase flow) why ruptured tubes in steam generators may cause breakdowns.

This leads to the following breakdown sequence:

As a result of the bursting of a tube in the steam generator, an excessive quantity of primary coolant enters the secondary loop and increases the pressur there as a result of which the discharge valve responds.

The discharge valve of a defective steam generator breaks down in the position of "open" or "partially open" (individual error).

This leads to the release of radioactive substances into the environment.

The Breakdown Sequence

- Breakdown of a tube in a steam generator (triggering process)
- Response of the secondary discharge valve of the defective steam generator.
- Breakdown of the secondary discharge valve in a position of "open" or "partially open."

(Individual error)

results from the application of the individual error criterion to the post heat discharge in coolant losses; the first two accidents actually happened at Ginna, U.S.

A more detailed description of the breakdown sequence as well as the computations of releases under different marginal conditions and their evaluation will be given below:

228

6.2. Description of the Breakdown Sequence

As a result of the breakdown of 10 tubes in the steam generator, the main coolant infiltrates the secondary loop. This leads after approximately 10 seconds to the rapid shutdown of the reactor by signal 'N16 > max' and after another 15 seconds to the delayed quick shutdown of the turbine.

At this point, that is approximately 25 seconds after the beginning of the breakdown, the pressure has increased to such a degree that activates the discharge valve.

In accordance with the individual error criterion, the breakdown of the discharge valve is assumed due to its "open" or "partially open" position.

Three further scenarios are established for the continued sequence of breakdowns. J 131 is chosen as guiding nuclide. As a result of the transient and spiking factors the interpretation radioactivity was established between 25 and 100.

Establishment of the Marginal Conditions Applicable to All Scenarios

- Only the radioactivity release during the first 30 minutes after the beginning of the accident was computed. This period of time is not considered to be conservative since subsequent rele

228

may occur.

\- About 1 minute after the beginning of the accident (it may differ according to the scenarios) the main coolant pumps are shut off by the signal "pressurizer water level < min". Since the signal "primary pressure < min" has responded (?), the flow into the loop begins through the safety feed-in pumps. In this connection, only two (out of four) feed-in pumps have been considered.

This is not an absolutely conservative assumption, since after consideration of the 3 or 4 high pressure feed-in pumps, the main coolant leakage increases in the seondary part of the defective steam generator.

\- During the first 25 seconds after the beginning of the accident, steam releases from the high-pressure part of 1 kg/sec and the feed water degasification of 0.5 kg/sec are assumed.

229 - For the purpose of determining the radio activity discharge through the discharge valves into the environment only the primary coolant quantity discharged from the discharge valve of the defective steam generator has been considered.

\- During the first 20 seconds after the beginning of the breakdown the average primary coolant leakage into the secondary loop (in the following to be called primary coolant leakage) is approximately 310 kg/sec.

The primarly coolant leakage drops to 280 kg/sec from 20 to 25 second

Afterwards the leakage quantities will vary according to the scenario.

<u>Marginal Conditions in Scenario 1</u>

A spiking factor of 25 for J 131 is assumed.

0 - 25 sec after the beginning of the breakdown

- average residual moisture in the steam generator 30%

25 - 300 sec after the beginning of the breakdown

- average residual moisture in the steam generator 20%
- primary coolant leakage dropped from 280 kg/sec to 80 kg/sec
- release of steam from the defective discharge valve (in "partially open"position) of the steam generator and its burst tubes (called steam release below) is on the average 120 kg/sec

300 - 1800 sec after the beginning of the breakdown

- average residual moisture in the steam generator 5%
- primary coolant leakage drops and averages 40kg/sec
- steam release drops and averages 20 kg/sec

Marginal Conditions in Scenario 3

A spiking factor of 50 for J 131 is assumed.

0 - 25 sec after beginning of breakdown

- average residual moisture in steam generator 30%

25 - 100 sec after beginning of breakdown 20%

- average residual moisture in steam generator 20%

- primary coolant leakage drops from 280 kg/sec to 140 kg/sec

- average steam release 150 kg/sec

100 - 1 800 sec after beginning of breakdown

- average residual moisture in steam generator 10%

- primary coolant leakage drops and averages 70 kg/sec

- steam release drops and averages 50 kg/sec

Marginal Conditions in Scenario 3

A spiking factor of 100 for J 131 is assumed.

0 - 25 sec after beginning of breakdown

- average residual moisture in steam generator 30%

25 - 1 800 sec after beginning of breakdown

230

- average residual moisture in steam generator 20%

- primary coolant leakage drops and averages 140 kg/sec

- steam release averages 150 kg/sec

It would be desirable to examine in greater detail the above and additional scenarios by means of further studies and possibly with the aid of measuring programs. In this connection, it would be especially important to ascertain how many tubes in the steam generator (with discharge valve in "c position would have to break down to bring about a meltdown accident in the In such a case, a portion of the core equipment might be released into the en as a result of which the releases in the abovementioned scenarios would multiply.

231

6.3 ~~[illegible]~~ Computation of the J 131 ~~[illegible]~~ Releases and Their Evaluation with Program SEKEM 4

The radio activity releases were calculated with Program SEKEM 4 under the marginal conditions indicated.

In the following Table 6.3.1., the releases and the release rates during the individual periods have been described in Scenario 2 (to be as an example). The releases refer to the periods indicated.

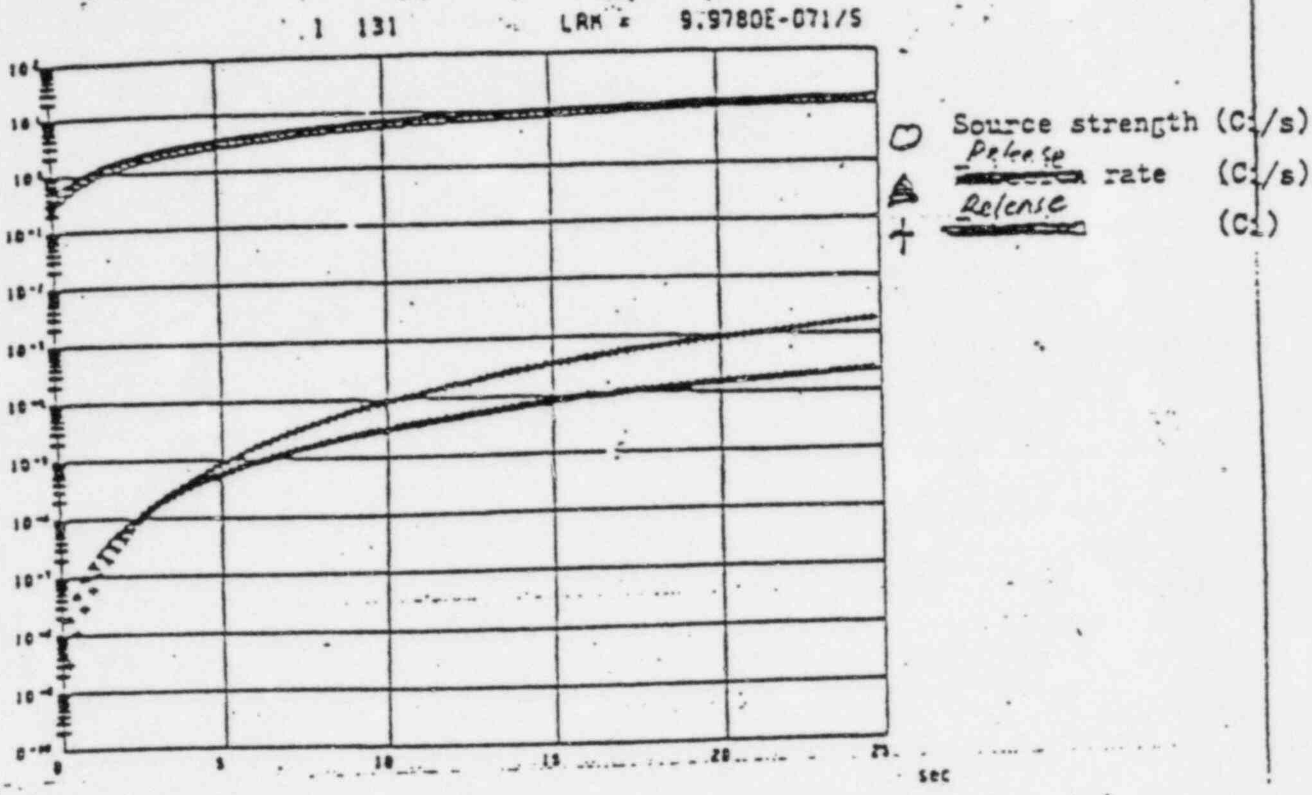

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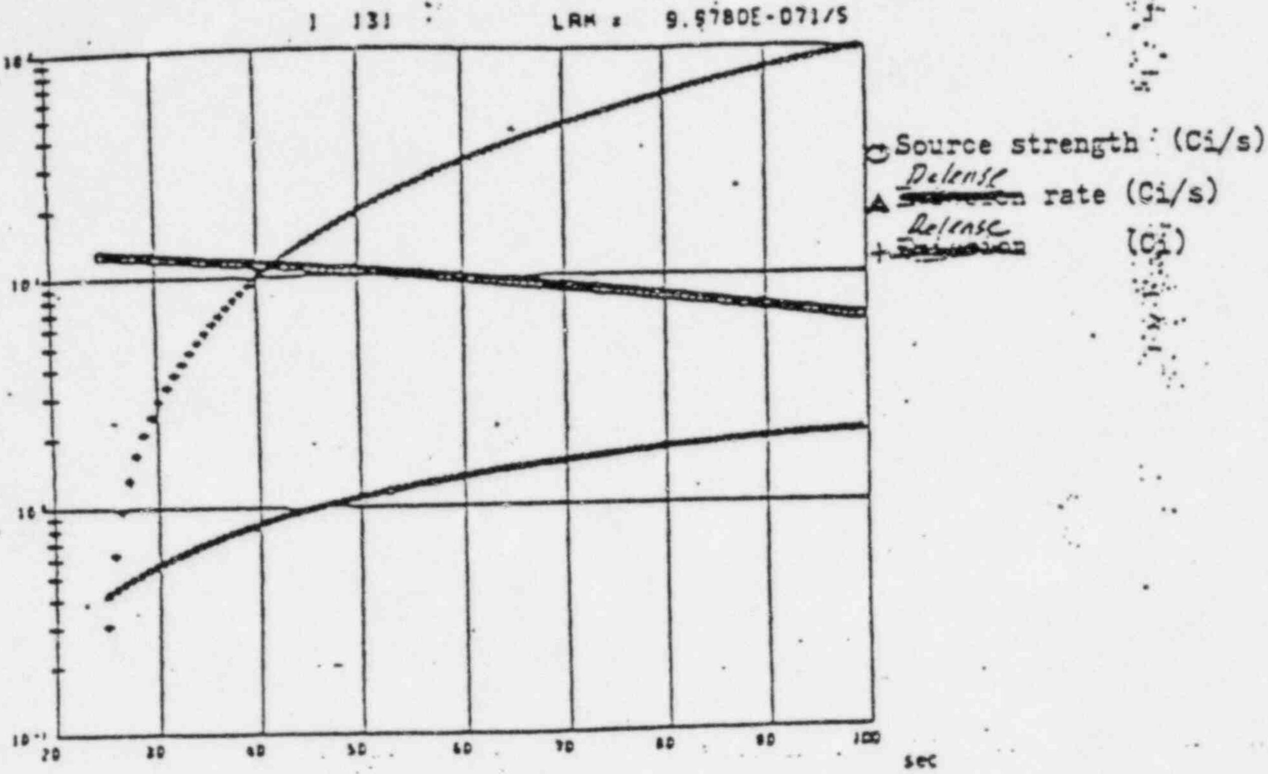
Table 6.3.1 Releases or release rates as result of the breakdown according to Scenario 2.
The source strength represents, as indicated, the product of time-dependent primary concentration and time-dependent leakage rate.



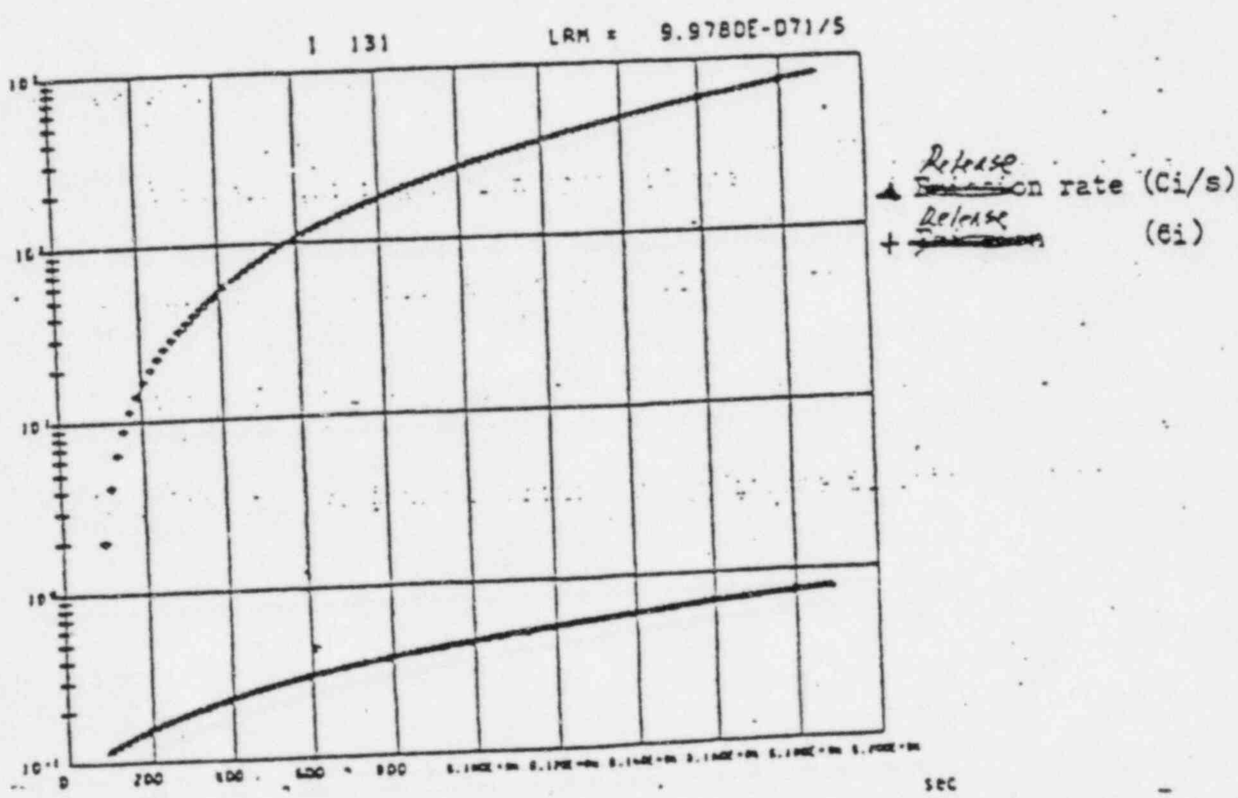


Table 6.3.1 Releases or release rates as a result of the breakdown according to Scenario 2. As indicated the source strength represents the product of time-dependent primary concentration and time-dependent leakage rate.

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233

In the following table the J 131 releases of the individual Scenarios during the first 30 minutes after the beginning of the breakdown have been listed.

In addition, the inhalation dose (only J 131) for adults and the thyroid gland of unborn children (fetuses) have been indicated under breakdown conditions.

Breakdown scenarios	J-131 ~~emissions~~ releases in 30 min	Inhalation dose for adults [rem]	Dose for fetal thyroid gland [rem]
Scenario 1	185	12	31 - 250
Scenario 2	830	54	140 - 1,100
Scenario 3	14400	940	2400 - 19800

Table 6.3.1 Release from J-131 in (Ci) and inhalation dose [rem]

Breakdown limit: 15 [rem].

Breakdown sequence: burst tube in steam generator and discharge valve breakdown in "open" or "partially open" position.

33

The table shows that already through inhalation the breakdown limit of 15 rem for unborn children (fetuses) has been exceeded in all cases and for adults in scenarios 2 and 3.

In scenario 2, the limit for adults exceeds around factor 3.6 and in scenarion 3 around factor 60.

The load of all other released nuclides must still be added to the above.

234 We proceeded from the assumption that in the event of such breakdowns the consumption of food from the contaminated regions will most likely be prohibited by the authorities.

For this reason only the inhalation dose of the released J 131 was calculated. If ingestion loads were included, the dose values would be much higher.

Calculation of the Breakdown Dose

The inhalation dose D under breakdown conditions is calculated as follows:

$$D = A \cdot X \cdot V \cdot D/Qi$$

with A = released radioactive nuclide quantity [Ci]

X = short-time propagation factor [s/m^3]

V = breathing rate [m^3/Ci]

D/Qi = inhalation factor [rem/Ci]

The following values were used:

- The released J 131 radio activities for the individual scenarios are listed in Table 6.3.1
- The Technical Control Board Hannover indicates a value of $2x10^{-4}$ (s/m³) at a distance of 150 m as short-time propagation factor for release from the turbine hall (6). In this connection, the upward pressure of the radi activity cloud caused by the large quantities of superheated steam that were simultaneously released has already been taken into consideration.
- As a inhalation rate the value for adults of 20 m^3/d ($\hat{=}$ 2,31 .$10^{-4} m^3/s$) was obtained from (7).
- Also the same source furnished the employed inhalation dose factor for the thyroid gland (J 131) of 1,4 . 10^6 rem/Ci.

As exhaustively explained by Steinhilber-Schwab and Franke (8), the fetal thyroid gland represents the "most unfavorable area of action via the load path.

In Table 6.3.2, the ratio between the fetal thyroid gland and the refer value according to (7) has been represented.

Accordingly, the fetal thyroid gland dose is higher than the reference value around factor 2.6 to 21.

	Dose factor (BMI 1979) (in mrem/pCi)	Inhalation rate	Dose in Jod 131 in air concentrations of 1 pCi/m^3 9in mrem/d)	relative thyroid gland dose, adults (BMI) = 1
Addlts	1.4×10^{-3}	20 m^3/6	2.8×10^{-2}	
Children	1.2×10^{-2}	2 m^3/6 (in position of rest (ICRP 23)	2.4×10^{-2}	0.85
		6 m^3/d (in activity ICRP 23)	7.2×10^{-2}	2.6
Fetuses	J 131 concentration in fetal thyroid gland 2 - 12 timesof that of adults according to BMI (1979)	27m^3/d (in easy activity ICRP 23)	7.5×10^{-2} bis 4.5×10^{-1}	2.6 - 16
		36 m^3/d (in increased activity ICRP	1.0×10^{-1} bis 6×10^{-1}	3.5 - 21

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Table 6.3.2. Comparison of the J - 131 inhlation dose of adults, small children and fetuses (according to Lit. (8))

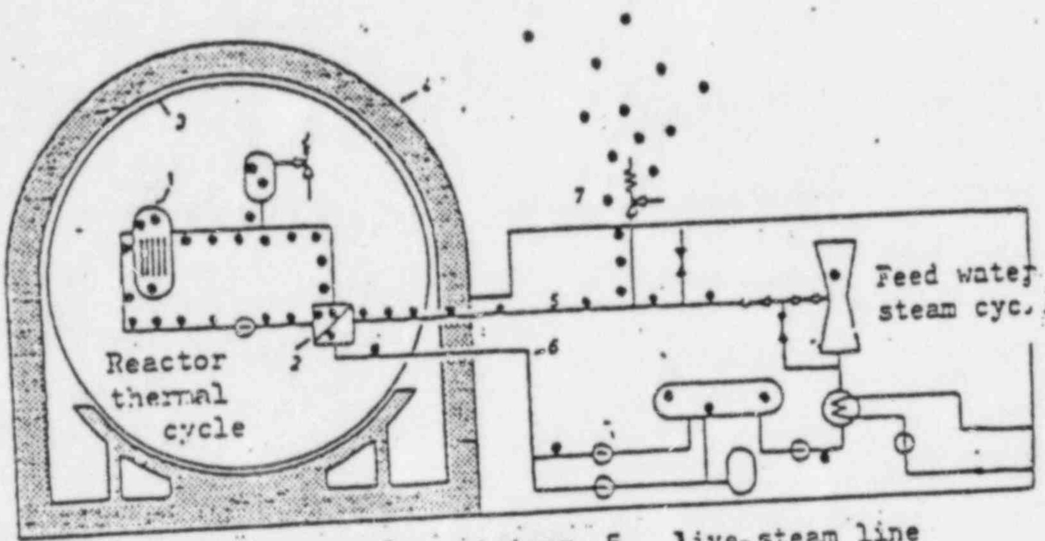
235 Concludingly it may be said that in view of the coolant loss breakdown and the applic tion of the individual error criterion the breakdown emission limits may be increased many times over; in adults the limit has alre dy been exceeded by factor 60 through the inhalation of J-131.

In the breakdown described: burst hot tube pipe in steam gen erator (triggering event), failure of the discharge valve in "open" position (indiviudal error) also accidents caused by core meltdown leading to consider ly higher release radio activities must not be excluded.

236 Special importance should be given to this sequence of breakdowns in future observations of risks. In addition, the breakdown must be considered to be an interpretation breakdown in atomic law authorization procedures.

Insidious in this breakdown is that radio activity via steam generation and discharge valve is released into the environment by bypassing all safety barriers (see Table 6.3.2.)

236



1. Reactor pressure container
2. Steam generator
3. Safety container
4. Reinforced concrete casing
5. Live-steam line
6. Feed water line
7. Discharge valves (Live steam safety valves and discharge exhaust valves).

o Activity

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Table 6.3.2. Release of (fallo) activity by bypassing all safety barriers Sequence of breakdown: Breakdown of tubes in steam generator- Exhaust valve in "open" position

6.4. A Critical Examination of the Breakdown Studies Carried out by the Baden Technical Control Board

In its examination "Release of Radioactivities in Atomic Plants Breakdowns with LWR [meaning of abbreviation unknown] Via the Exhaust Path Within the Framework of a Guideline to Paragraph 3, Article 28 (as of December 14, 1979)" the Baden Technical Control Board likewise consid breakdowns with secondary releases (through roof of turbine hall). For this purpo the KKP II (Philippsburg, Baden) installation now under construction has served for the examination of the KWU-pressurized water line.

From the list of the breakdowns discussed in the above examination two interesting breakdowns are discussed in greater detail in connection with the abovementioned examination. They are:

1. Breakdown of a live-steam line outside of the safety container coinciding with the breakdown of 10 tubes in the steam generator.

2. Breakdown in the feed water system outside of the safety container coinciding with the breakdown of a tube in the steam generator.

No opportunity existed for the review the KWU programs modified by the Baden Technical Control Board and the system analysis based on them. For this reason the correctness of the calculation programs and computer

237 plots with regard to the algorithm and also the system analysis must be

assumed a priori.

Therefore consideration may be given to the sequence of breakdowns

only as far as those criteria are concerned that are particularly

relevant to releases. Special attention should also be given here to

radio activity production data.

6.4.1. Breakdown 1 - Bursting of a live-steam line outside of the

safety container coinciding with the breakdown of 10 tubes in

the steam generator.

The Technical Control Board computes the quantity of activity

released in 30 minutes as a result of a breakdown in the live-ste

line coinciding with the breakdown (10 tubes) in the steam generator at

238 the beginning of the breakdown in the live-steam line leakage and, subsequent

via the discharge valves of the steam generator via the turbine hall.

The breakdown has not been chosen conservatively since only a breakdo

of 10 tubes in the steam generator has been assumed. Altogether there are more

than 4,000 U-shaped tubes in a steam generator. According to our

findings in transients the breakdown of at least 20 tubes should be assumed,

38

The release results from the primary cooling water released into the environment and its specific activity.

In its computations, the Technical Control Board proceeds from a release of primary coolant of 191.63 kg with a specific (radio)activity of 21.6 Ci/t with a Jod-131-share of 0.92 Ci/t.

Both specifications must be challenged.

1. The primary coolant release of approximately 200 kg is too low by factor 27 as it can be recalculated from the Technical Control Board's own data and assumptions. In addition correction of the untenable assumptions about the residual moisture in the steam generator has not been made.

2. the specific activity of the primary coolant of 21.6 Ci/t indicated by the Technical Control Board is too low at least around factor 4.6 in average spikes and 19.3 in maximal spikes.

This specific activity of the primary cooling water may already be exceeded over short periods in normal operation as a result of load changes (small transients). This spiking effect grows with the size of the transients. A maximal transgression of the interpretation activities is, therefore, to be expected, particularly, in assumed breakdown conditions with large pressure transients. Therefore, the spiking factors must be added [!] to the

238

interpretation activities (see Section 2.3.1.2.).

In addition to the spike criterion, consideration must be give

239

to the fact that Tables 3.22 and 3.23 of the Technical Control Boar which all computations of the activity release refer, do not include all relevant nuc and that the data always exclude trit activities.

In connection with the breakdown discussed here - breakdown in the live-steam line coinciding with the bursting of tubes in the steam generat it is suggested that we examine here this breakdown and the additional breakdown of a discharge valve in "open" or "partially open" position in agreement with the individual error criterion.

By doing so, we would have to consider the breakdown of the tube as an error resulting from the bursting of the live-steam line. In this breakdown significantly higher releases may be expected as a result of t breakdown of the discharge valve.

6.4.2. Breakdown 2- Breakdown in the Feed Water System Coinciding the Breakdown of a Tube in the Steam Generator

After the breakdown of the feed water line at the lowest point of the feed water system, its contents empties together with the live-steam portion into the roof of the ~~turbine hall~~ turbine hall (altogether 730 t) during which time part of it evaporizes (apprx 170 tons) and is released into the environment via the roof of the turbine hall.

88 kg of water with primary activity infiltrates the feeder via the turbine hall sump system. According to the Technical Control Board these 88 kg infiltrates with the live steam tube leakage the feed water container via the condenser by the time the turbine is shut down (30 sec).

In this connection the following may be stated:

1.. The breakdown sequence contains a few conservative assumptions:

Only one single tube breakdown has been assumed.

Already at the burst of 2 tubes, the response pressure of the discharge valves (86 bar for KKP II) would be exceeded and a significant additional

240 release would result (see breakdown 1 - breakdown of the live steam line).

As described in Section 2.4, a sudden breakdown of tubes in the steam generator must be assumed (and in particular, the breakdown of at least 10 tubes in transients subject to breakdowns. In addition, we must furthermore assume that not all tube leakages occur in one steam generator.

2. The computation of the activity release is contradictory.

2.1 It is assumed that 88 kg of primary water are released through the feed water leakage and that part of it evaporizes together with the 170 t and is released through the roof.

If we proceed from the assumption that the 88 kg must be equally distributed over the 730 t of the total water leakage, 20.5 kg of primary water would have to evaporize with it. In this connection, it is assumed that evaporization quantity of 170t has been correctly computed. The Technical Control Board did not take into considerati the activity adequate for this primary coolant quantity of at least 2.1 Ci in average and 8.5 Ci in maximum spikes.

2.2. The formula used for the computation of the live-steam radio activity in operational leakages is not sufficient. The formula does not consider the radio activity return flow from the feed water container into the steam generator (as shown in Chapter 4 - sensitivity analysis - this return flow is primarily for the-live steam concentration).

2.3 The radio activities indicated in Table 3.22 show discrepancies in tho:

240 values ~~[illegible]~~ calculated from the formula indicated. A comparison between ~~of~~ the live steam values with the data in Table 3.23 "specific nuclide activity ~~[illegible]~~ in steam generator water" indicates that the Technical Control Board, on the basis of ~~[illegible]~~ the current status of science and technique, has consistently employed excessivel high ~~[illegible]~~ decontamination factors which results in an underevaluation of the ~~[illegible]~~ release.

241 Concludingly, it may be stated that the sequences of breakdown ~~sequence~~ in both examples examined are ~~is~~ not conservative in many points which leads to an under evaluation of the releases ~~[illegible]~~. This applies, in particular to the number of burst ~~[illegible]~~ tubes in steam generators and the primary loop ~~[illegible]~~ activities. If the abovementioned factors are taken into consideration, the breakdown li may be exceeded by J-131.

242

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243

7. CONCLUDING REMARKS

7.1 Consequences Derived ~~from~~ the Results of the Study

The computations of releases ~~in~~ operational leakages in steam generator tube and breakdowns in steam generators show that the primary loop concentrations represent central input parameters,

In order to minimize in normal operation the risks connected with an increase of the primary loop radio activities, we propose to expand in terms of the primary circuit the reactor protective system by means of a measuring device for the purpose of continuously monitoring the radio activity concentrations of some guiding nuclides. After exceeding a certain limit the reactor should shut off automatically (see Lit. (5) and (1). In (1), Osetek and others present a special γ- spectrometer with which the fission productivity of the primary coolant can be measured continuously

The burst tubes in the steam generator after examination show that the corresponding secondary releases have so far been entirely underevaluated:

If, by applying of the individual error criterium to a burst

243

tube in a steam generator, an additional breakdown of the secondary discharge valve in an "open" or "partially open" position is taken into consideration, the computations show an transgressior of the breakdown limit through inhalation already for Jod-131 by the factor 60 in the case of adults.

These results relate to the statment made in the "German Study of Risk" (3):

" A leak in the pressure holder or the steam generator itself is controlled by the safety systems just as a leak is in the primarly coolant line."

This statement does not apply to the breakdown described above which, in application of the individual error system, we propose to include as an interpretation breakdown in the series of the breakdowns to be considered within the framework of a guideline to Paragraph 3, Article 28 of the StrSchV, since a spontaneous breakdown of tubes in the steam genera-

244 tor may not be excluded even without large transients (see the mishap at GINNA, U.S . on February 1982).

In connection with the burst tube in the steam gener tor and breakdown of the secondary discharge valve in an "open" position a core meltdown accident as a result of coolant loss, cannot be excluded.

This would result in the release of a part of the nuclear ~~[illegible]~~ equipment via the secondary discharge valve. ~~[illegible]~~ The effect ~~this~~ it may have on the environment has already been examined by the Insitute for Reactor Safety (IRS) (2).

In order to minimize the catastrophic consequences of such a breakdown (the term "breakdown" has been purposely chosen since the sequence of breakdowns ~~[illegible]~~ results from the application of the individual error criterion to the ~~breakdown~~ coolant loss breakdowns), we propose ~~to~~ to build ~~[illegible]~~ a containment for the purpose of preventing corresponding releases from the secondary discharge valve from ~~[illegible]~~ infiltrating the environment. In this connection, we also wish to mention ~~[illegible]~~ the construction of a subterranean safety tunnel through which the exhaust valves ~~[illegible]~~ will release steam (4).

In addition, the following consequences should be drawn from the results of the study:

- The air conditiong system in the switch plant for the control room, etc. gets is air humidity (ca. 160 kg/h) from the auxiliary steam system. The auxiliary steam system, however, will in accordance with the operational condition of the power plant be supplied either ~~[illegible]~~ by the high pressure tap system, the ~~[illegible]~~ live steam system or ~~[illegible]~~ the auxiliary steam generator system. This should be changed immediately since in the event of a burst tube in the steam generator the radio activity that infiltrated the live-steam is distributed over the air conditioning system.

The shut-off limit of the N-16 signal should be coupled to the reactor output since in a 50% output as compared to a 100% output the steam generator output might be approximately twice as high prior to the N-16 quick shut-off's response.

- Particularly, in the steam generator, a direct steam transport should be taken into consideration in most nuclides in addition to the residual moisture transport, since both live steam radio activity and also releases might be undervaluated.

- For compliance with the maximum live steam concentration (for J-131) as established by the authorities should by continuous be measured. Otherwise a transgress: might occur.

- Turbine hall releases should be monitored by measuring instruments. As long as this is not the case, repeated differences measurements of releases and immission quantities similar to those that happen

245

in in Obrigheim cannot be excluded since unmeasured activity releases may occur.

246

7.2. Bibliography to 7.

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2. The KWU Model does not assume any elutration demineralization release, and hence suppresses essential radioactivity contributing to a direct unfiltered release through the roof of the turbine hall.

3. The KWU Model assumes that the only direct release pathway is by a blow-off through the safety valve. It does not take into account releases from the high pressure part of the secondary loop. Both assumptions lead to a considerable understiamtion of the direct and unfiltered releases through the roof stack in normal operation (no blow-off).

4. The diagram suggests leakages into the turbine hall (with the exception of the blow-off through the safety valves). In contradiction to this is the assumption made on the completely false premisses in the text that although iodine-131 is released through the roof of the turbine hall (and as well as through the sewage), the N-16-scram limits the concentration of iodine -131 in live steam to the values decreed by the Federal Minister for the Interior. This point will be furter discussed in the chapter on the steam generator leakages.

We can briefly say, that the KWU Model must lead to a strong underestimation of real ~~bed~~ releases.

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