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Studsvik Report

IGNALINA RBMK-1500 BUILDING CAPABILITY IN RETAINING RADIOACTIVE RELEASES

Project IBBA, Stage 2. Basis for discussion of mitigation measures founded on natural convection principles

SKI Project No. 1.533 - 920030

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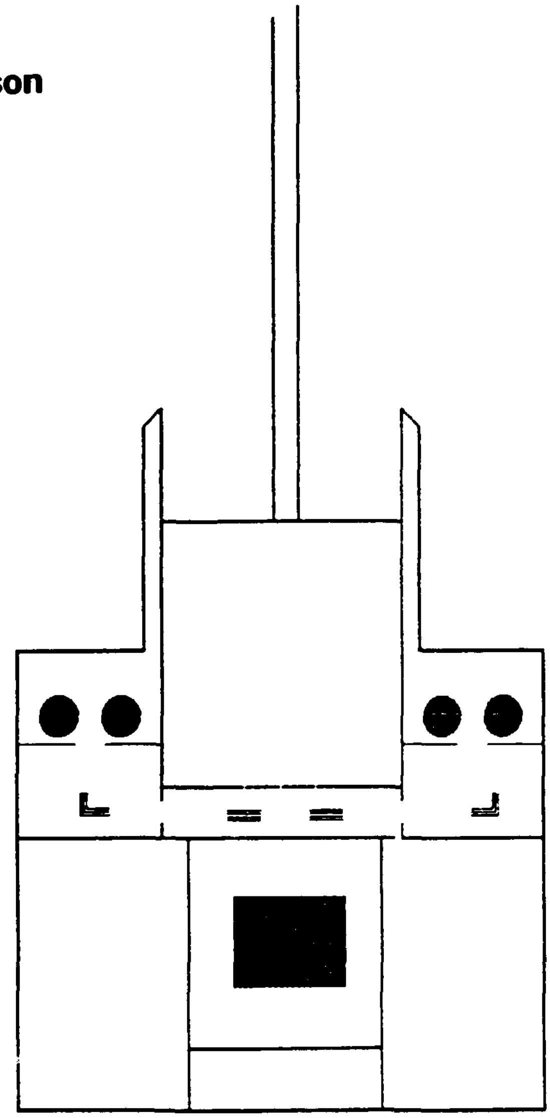

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Ignalina RBMK-1500 building capability in retaining radioactive releases.

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Abstract

The IGNALINA reactor building structures are capable of retaining substantial fractions of radioactive emissions from the fuel core, in those accident sequences where pressurization failure of structures can be averted by pressure relief arrangements. In stage 1 of the IBBA project it was demonstrated that enhanced retention of radioactive fission products within the plant can be achieved if natural convection is facilitated in the upper building compartments.

In this report of stage 2 is discussed for which accident sequences the introduction of natural convection in combination with the existing forced convection ventilation and the Accident Localization System can improve the the total safety of Ignalina 1-2. The purpose of this stage is to provide a basis for further review and more detailed studies of the natural convection concept, its benefits and disadvantages, and of the feasability to introduce the concept in existing plants.

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1 Introduction

1.1 Background

In the first stage of present project "IBBA" (Ignalina Building BArrier) [Ref 1] a tentative study was carried out of the capability of the Ignalina building to retain radioactive releases after introduction of natural convection in the upper compartments. The idea that the Ignalina building could be capable of retaining substantial fractions of radioactive emissions from the fuel core in certain accident sequences was based on experience from the Swedish research programs FILTRA and RAMA which have led to adoption of important mitigation measures in Swedish nuclear power plants.

The report from stage 1 of the IBBA project has been presented and reviewed by SKI and GRS. At a following meeting at GRS in Cologne [Ref 2], where also Studsvik EcoSafe took part, valuable view points came up and recommendations were made for future work. Stage 2, presented here forms an introductory step towards more detailed studies of the IBBA concept.

1.2 Purpose

The purpose of present work is to provide in broad terms a platform for discussions on the additional radioactivity retention that could be gained by making use of natural passive convection and aerosol removal processes. We expect interested participants in the discussion will in the first round be SKI, GRS, VATESI, Ignalina staff, LEI, and Studsvik EcoSafe.

The purpose of the early discussions will be to conclude on the desirability of further work on the lines indicated in this report and its supporting documents.

The purpose of later discussions will be to perceive of practical arrangements that could possibly be introduced and on a programme of work in support of evaluation of perceived measures.

1.3 Scope and outline of stage 2 of IBBA

The scope of this stage of IBBA is to make an inventory of possible accident sequences in which the concept of natural convection might have effect on release of fission products. Moreover, proposals will be made for feasibility studies and for following more detailed analyses of the retention capabilities compared to the existing design of the ventilation system.

The intention of stage 2 is to discuss in broad terms the radioactivity retention capabilities of the Ignalina building structures. It will in particular describe (our perception of) the function of the the steam pipe and drum compartment (zone 4, see Fig 1 and 6) and how its active ventilation and filtration system will perform in a few accident sequences.

The accident sequences will be chosen to represent events:

- inside the (zone 4) design basis
- somewhat beyond,
- clearly beyond and
- extreme events far beyond the design basis.

For the events chosen we will discuss the additional retention of radioactivity that could be contributed by passive natural convection and aerosol removal processes.

We intend to discuss how these passive natural removal processes could be enhanced by arranging for ventilation channels to make use of warm steam and gas to create a strong updraught that could be used to provide additional passive aerosol removal to complement the existing active ventilation and filtration system functions.

Introductory parametric studies of passive natural convection and removal processes have been performed as reported in [Ref 1].

A dedicated cooperative effort between Lithuania, Russia and Sweden in the Barselina project [Ref 3] has contributed towards creating among the participants a common perspective on the RBMK Risk Topography.

The results can serve, (as demonstrated here) as a framework for dicussing potential merits and possible drawbacks of passive natural air convection and aerosol removal processes applied to complement existing forced ventilation and filtration systems for reduction of accidental releases of radioactive aerosols at Ignalina.

The following *screening criteria* to select projected arrangements will be applied in choosing between several propositions, i. e. projected measures:

- should not lower safety within the existing design basis.
- should offer some benefit to safety in accident states somewhat beyond the design basis.

- should offer significant safety benefit in accident states clearly beyond the design basis.

- should add tools and capabilities to the operators repertoire to handle extreme events.

- should at least not deteriorate extreme accident situations which are beyond mitigation by accident management efforts. (e. g. Chernobyl type accidents).

2 Ignalina confinement system and its Design Basis

2.1 Brief plant description

Only a limited effort has been made within this stage of the IBBA project to acquire detailed data for the Ignalina 1-2 plants. Available translations are used here describing parts of the facility, such as selections of the RBMK-1500 Technical Safety Report [Ref 4] and other relevant reports [Refs 5 - 7] produced by RDIPE* and by IAEA [Ref 8]. A useful summary description made within the internordic SIK-3 project is found in [Ref 9], from which the main flow diagram (Figure 1) and the ECCS flow diagram (Figure 2) are copied. A simplified scheme of central parts of the plant is shown in Figure 3. As illustrated by Figures 1 - 4 the primary coolant system and its enclosing structures are divided into two halves arranged symmetrically on both sides of a vertically axial plane of the reactor core.

The RBMK-1500 building in Ignalina is designed with a somewhat different philosophy regarding safety and mitigation measures against radioactive releases compared to the concept used in western LWRs. In western PWRs and BWRs the containment forms a third barrier against radioactive releases to the environment, enclosing the reactor vessel and connected piping which can be isolated in case of a pipe rupture or a severe accident.

2.1.1 "Primary containment"

The Ignalina plant has a containment-like system, called the Accident Localization System, ALS, shown in Figure 4. The ALS encloses most of the primary system such as the main circulation pums, the reactor coolant inlet piping and some emergency cooling equipment in leak-tight cubicles. These cubicles are designed for an overpressure of 3 bar. Belonging to the ALS are also the ALS condensation towers with five storeys of condensation pools on each side of the reactor.

At a certain overpressure in the ALS cubicles, e.g. in case of a pipe break, the atmosphere is discharged by the steam distribution corridor into the four lower condensation pools of the ALS towers. There is also a large delay volume in the ALS towers to take care of non-condensable gases and letting radioactive noble gases decay before releasing them via a filter through the stack to the environment. During an initial time period, however, non-consable gases, mostly air, are released to the atmosphere until a specially designed ball valve closes the gas outlet. After this time, the atmosphere consists mostly of steam which is condensed in the pools and by means of a spray system.

* RDIPE = Research and Development Institute of Power Engineering (Moscow)

The reactor cavity with its graphite blocks is separately encased in a steel lining. In case of rupture of a zircaloy Pressure Tube (PT) in the core overpressurization of the cavity is limited by release to the ALS through special vent pipes which lead the primary system steam into the uppermost pools in the ALS towers. (The uppermost pools are also used for condensation of steam from the primary system at pressure relief) The capacity of the vent system allows for rupture of 3 PTs in the original design. After increasing the total pipe area the blow-off capacity should be enough for 11 PT breaks without violating the strength of the cavity lining [Ref 6]. If the cavity pressure exceeds 3.1 bar (absolute) the cavity top plate is calculated to be uplifted.

The ALS towers are also equipped with recombination facilities to reduce the hydrogen fraction in the atmosphere safely below its ignition limits during normal operation. This function, having a limited capacity, is, however, shut off in emergency situations.

2.1.2 "Secondary containment" and ventilation system

The exit pipes, connecting the pressure tubes above the top plate to the four steam drum separators, the steam drum separators with steam lines to turbines and the upper part of the downcomer pipes from the separators are placed outside the confinement of the ALS. These parts are located in a "secondary containment" and there are no isolation valves at the penetrations from primary to secondary containment (Figures 3 and 4).

The secondary containment is not isolated from the ambient atmosphere. There are also rather large leak areas, totally about 5 m^2, in the gaps around the shield blocks in the reactor hall floor, and the walls of the reactor hall are not leak-tight. The shield blocks have to be removable to facilitate access to the core channels, e.g. at refueling. A forced flow ventilation system, shown in principle in Figure 5, keeps an underpressure in the secondary containment. Fresh air is sucked into the reactor hall, passes the shield blocks into the steam drum compartment and the atmosphere is then blown through a filter to the outside via the 150 m high ventilation stack. The forced flow ventilation system is designed to take care of normal, small leaks from the primary system. The fan flow to the stack is about 13 m^3/s.

2.2 Design Basis and other anticipated accidents

Accidental pressurization of confinement compartments can be caused by steam blowdown of primary coolant water and or by hydrogen burn.

For the ALS compartments, the maximum Design Basis Accident (DBA) is a guillotine break in one of the 900 mm I.D. suction headers downstream the main circulation pumps and assuming simultaneous failure of a check valve upstream one of the group distribution headers, of which there are twenty on each half of the reactor. This transient has been analyzed and is described in [Ref 4].

The DBA and other smaller sized pipe breaks inside the ALS compartments, as well as rupture of up to eleven pressure tubes in the core should be within design and be taken care of by the ALS and the emergency cooling systems (ECCS) without violating the integrity of the primary containment or other parts of the building. The ECCS water is supplied to the primary system piping at the inlet side of the core. In case of pipe rupture in an inlet pipe, connected core channels are cooled by fluid from the steam drum separator from above the core.

Breaks or leaks in primary system components in the secondary containment have been considered to be less harmful, based on risks of release of radioactivity to the environment. Such steam losses are believed not to lead to fuel cladding failures. However, any detailed analyses of these kind of accidents are not known. It can not be excluded that there are scenarios in which release of fission products may occur through steam leaks outside of the ALS, e.g:

- steam leaks from fuel channels with earlier damaged claddings. Early cladding failures are not unlikely due to the rather small critical power ratio in the RBMK-1500 fuel bundles even at the currently reduced power of 1300 MW_e in Ignalina 1-2.

- flow blockage to one or several pressure tubes in the core might lead to overheating of upper parts of the PTs and possibly also of their connections to the steam drums which are outside the ALS. These pipe ducts have a large number of welds with concomitant risks for ruptures at elevated temperatures.

- a large break in a steam drum separator or of one of its 287 mm I.D. steam outlet pipes with simultaneous total loss of electric power will probably lead to severe core damages with release of large fractions of fission products.

- rupture of more than eleven PTs exceeding the design limit of the ALS. The pressure in the reactor vault will then lift the top lid which in turn can lead to damage of a large number of steam outlet pipes in the secondary containment.

These and other anticipated accidents should be analysed using deterministic methods.

3 Confinement zones

Five pipe rupture location zones are defined according to [Ref 3]. The zone borders are marked out in Figure 6.

3.1 Zone 1

Inside the primary circuit confinement upstream the Group Distribution Header (GDH).

The ALS is capable of relieving and condensing steam from coolant blowdown at the rates occurring in the event of a 900 mm guillotine break in a main coolant pump discharge header (O.D./I.D. = 1040/900 mm).

3.2 Zone 2

Inside the primary circuit confinement downstream the GDH check valve.

The Accident Localization System is capable of relieving and condensing steam from coolant blowdown at the rates occurring in the event of rupture of a Group Distribution Header downstream of the check valve of roughly 0,3 m diameter (325/295 mm).

3.3 Zone 3

Inside the reactor cavity.

The ALS is capable of relieving and condensing steam from coolant blowdown at the rates occurring in the event of simultaneous ruptures of 3 pressure tubes according to the original design.

After planned modification, the steam relief capacity of zone 3 will be increased corresponding to multiple blowdown of up to 11 pressure tubes.

For multiple ruptures in excess of 11 the cavity vessel could rupture and relieve steam either upwards into zone 4 or downwards into the ALS.

For very large numbers of multiple pressure tube ruptures the cavity cover plate structure could lift and cause additional pressure tube ruptures and uncover the core.

3.4 Zone 4

Outside the primary circuit confinement on the primary circuit side of isolation check valves.

Zone 4 contains the pipe conduit and steam drum compartments on top of the core cavity. They are in the secondary containment, i. e. outside the ALS.

Steam from coolant blowdown in the event of rupture of a steam drum outlet pipe (325/287 mm), a main steam line (630/580 mm) or feed water line (325/293 mm)) will be relieved directly to the atmosphere via nine rupture panels. They open at an overpressure of 0,02 bar and have a total relief flow area of 20 m^2.

Steam from rupture of the individual core exit pipes (76/68 mm), carrying coolant from the fuel channels to the steam drums, will be relieved the same way.

Also steam from upward rupture of zone 3 will be relieved via zone 4 in this manner.

3.5 Zone 5

Zone 5 is outside zones 1 - 4 defined above, i. e. outside the primary circuit confinement on the secondary side or interfacing system side of isolation check valves.

Some pipe ruptures (also in e.g. the steam lines) can occur on the secondary or interfacing systems side of isolation check valves outside the zones 1- 4 defined above.

An unisolated rupture in zone 5 will become a zone 4 event [Ref 3] (because of internal pressure relief paths between the building compartments).

A particular case is rupture of a pressure tube at the top such that release of steam and radioactivity occurs into the reactor hall. This type of release situation is outside our scope and is not further discussed in this report.

Zone 5 pipe ruptures can to some extent be included under the natural circulation and removal strategy we have in mind here. How and to what extent will be subject of the future study.

4 Accident initiators

The accident initiators discussed in [Ref 3] are large, medium, small LOCAs, transients and Primary Circuit Blockages (PCB) of coolant flow in one or several pressure tube fuel channels.

The PCBs are prominent contributors to the frequencies of events in all four groups of core damage states discussed below.

Transients are prominent contributors to the clearly beyond, and the LOCAs to the somewhat beyond the Design Basis.

The LOCAs also initate the (Zone 4) Design Basis Events.

5 Core and confinement states

For the purpose of the following discussion we intend to define accident initiators, core damage states and confinement damage states in each of four groups representing the following events:

- inside the (zone 4) design basis
- somewhat beyond,
- clearly beyond and
- far beyond the design basis.

As a basis for a discussion of plant states and possible source terms we suggest a classification under the following headings, which are also employed in Table 1:

A Core States

A.1 Core damage states inside the (zone 4) design basis.

A.1.1 Rupture of pipe (in zone 4) between top of fuel channel and steam drum

A1.2 Rupture in (zone 4) of a main steam line.

A.2 Core damage state, somewhat beyond the (zone 4) design basis.

A.2.1 PCB, Primary Circuit Blockage

A.3 Core damage state, clearly beyond the (zone 4) design basis.

A.3.1 Blockage of Group Distribution Header causing large number (44) of pressure tube breaks in core cavity

A.4 Core damage state, far beyond the (zone 4) design basis.

A.4.1 Station Blackout, extended in time

A.4.2 Rupture of main steam line in combination with total loss of electric power

A.4.3 Rupture of main coolant pump discharge header in combination with total loss of electric power

A.5 Extreme event: Reactivity excursion of the Chernobyl type

B Confinement States

B.1 Confinement (Zone 4) states

B.1.1 Confinement boundary leaktight. Active ventilation and filtration available.

B.1.2 Confinement relief paths open to the atmosphere.

B.1.3 Confinement relief paths reclosed. Passive natural convection and aerosol removal.

B.1.4 Confinement boundary failed (by pressurization beyond relief capability).

6 Hydrogen mitigation strategies

A study, following upon this stage 2 of the IBBA project, should recognize the potential impact of hydrogen burns on building structures integrity.

There are three main hydrogen mitigation strategies, namely gas inerting, steam inerting and passive recombination. Also active recombination and other means of hydrogen mitigation can be conceived of during the course of the study.

The purpose of next step in the study is to investigate how, and in what accident states natural and forced ventilation in concert with passive or active hydrogen recombination can contribute towards safeguarding of building structures and reduction of radiactive releases.

7 Conclusions and suggestions for future work

The effect of introducing natural circulation in the uppermost part of the Ignalina building as a means to reduce release of radioactivity after a severe accident was studied in stage 1 of the IBBA project [Ref 1]. The concept is based on utilizing the driving force from heat losses from the large uninsulated surfaces of the steam exit components, which heats up the atmosphere in the secondary containment above the core, and use of the draught in the 150 m high ventilation stack. A substantial retention of fission products as aerosols can be achieved, provided that leak areas in the floor of the reactor hall or inleakage of air from the outside can be limited and controlled .

The existing ALS system is designed to mitigate accidents within the primary containment, i. c. DBA- and smaller-sized LOCAs in the primary system, so that releases are below regulatory limits. In present report of IBBA, stage 2, is pointed at anticipated accident sequences, mainly caused by primary system breaks in the secondary containment, in zone 4 and 5, which are not taken care of by the ALS. Core damage states and confinement states are listed for these cases. Accidents leading to strong overpressurization and rupture of the reactor vault pressure boundary, such as events of the Chernobyl type are not considered.

Table 1 shows some tentative figures of relative values of source terms for Ignalina as now operated, based on engineering judgements only. L. Vecsler of Kurchatov Institute gives some values of source terms and a classification of plant damage states that is referred to in the Barselina project [Ref 3, chapter 4.3]. How the figures by Vecsler are arrived at is , however, not known.

In next stages of the IBBA project, which might follow, further studies of existing analyses on source terms should be made and, if necessary, own calculations should be performed for the Ignalina plant as built. This calls for access to more detailed station data regarding safety system, building measurements, flow paths, etc. In order to assess the the efficiency and the benefits of the IBBA concept comparative analysis has to be made by means of deterministic methods.

We recommend the following future steps:

a) present the IBBA concept to Ignalina staff and concerned authorities for discussion of its benefits, possible disadvantages and its feasability with respect to practical arrangements, e. g. possibilities to combine it with the existing forced flow ventilation system.

Given acceptance to proceed with further work on the IBBA concept, we recommend:

b) retrieve more detailed information on the Ignalina plant with emphasis on data of present forced ventilation system of importance for analyses of the building inherent aerosol retention capability with existing design.

c) select one or a few of the zone 4-type accident sequences discussed above and calculate, e. g. by means of CONTAIN, source terms for the existing design of the ventilation system.

d) make a feasibility study of the possibilities to introduce the IBBA concept of natural convection in combination with existing ventilation system. Investigate necessary practical arrangements and see how to incorporate them into the present building. Special attention should be paid to how to reduce and control the gas flow areas in the reactor hall.

e) investigate how to combine the two ventilation functions (forced and natural convection). If any kind of switch-over from one system to the other is necessary, suggest ways to implement this.

f) calculate the natural convection source terms for the same accident scenarios which were studied with the original forced flow ventilation design. Compare and discuss the results.

g) investigate the interaction between the ventilation system (forced and/or natural convection) and hydrogen recombination method (passive or active) and how this can influence the reduction of radioactive releases.

For the first approach, postulated release fractions from the primary system will be employed, instead of calculating them all way from the initiating event.

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Table 1 **Over view of Plant, Core and Containment States with their corresponding source terms**

INES	PLANT STATES		CORE STATES				CONTAINMENT STATES			SOURCE TERMS	REMARKS
	ACCIDENT INITIATORS	SUPPORT & SAFETY FUNCTIONS	FUEL STATES	PRESSURE TUBE STATES	HYDROGEN STATES	GRAPHITE STACK STATES	ZONE 3 STATES	ZONE 4 STATES	ALS STATES	As Now Operated	
0			S= Safe stable cooling			Operating temp C	Inerted	Inerted ?	Inerted ?	$S = e \cdot L^*$	
1 2 3	LOCAs Transients	Available apart from single failures	V= Violation of heat remov from fuel at preserved geometry $P = 5{,}7 \cdot 10^{-3}$	Small local damage	Small amounts	Heated locally				$e = 10^{-5} \rightarrow 10^{-3}$ $L = 0{,}01$ $S = 10^{-7} \rightarrow 10^{-5}$	Inside Design Basis
4	Local flow blockages Station blackout	temporary	D= Damage to fuel by local melting $P = 3{,}2 \cdot 10^{-4}$	Multiple press tube rupture at low press		Graphite stack cooled via control rod coolant flow	Fuel channel top end heat up to failure into Zone 4 Low end fuel meltout into ALS	Structure of Zone 4 remain sound	Molten fuel opens vent path via Zone 3 to Zone 4	$e = 0{,}002$ $L = 0{,}1$ $S = 0{,}0002$	Somewhat beyond Design Basis
5	Station blackout	longer	A= Severe Accident $P = 1{,}4 \cdot 10^{-6}$		H2 evol from fuel and press tubes	Control rod channel boil dry and graphite heat up				$e = 0{,}1 \rightarrow 0{,}5$ $L = 0{,}1 \rightarrow 0{,}2$ $S = 0{,}01$	Clearly beyond Design Basis
6	Station blackout	prolonged		Multiple press tube rupture at high press		Air ingress to hot graphite causes fire	Upper core plate uprise into Zone 4 Low end fuel meltout into ALS	Zone4 structure failed by press beyond relief Transit of FP+H2+steam via Zone 4 to atmosphere	Molten fuel opens vent path via Zone 3 to Zone 4	$e = 0{,}5 \rightarrow 0{,}8$ $L = 0{,}2 \rightarrow 1$ $S = 0{,}1$	Severe Accident far beyond Design Basis
7			RIA induced power transient eg Chernobyl	-"-	?	H2 and graphite fires	Open upwards to zone 4	Open to the atmosphere		Kr Xe S=1,0 131I S=0,2 137Cs S=0,13	Extreme event

* e = fraction of Cs and I inventory emitted from fuel, L = fraction of emitted leaked to the atmosphere

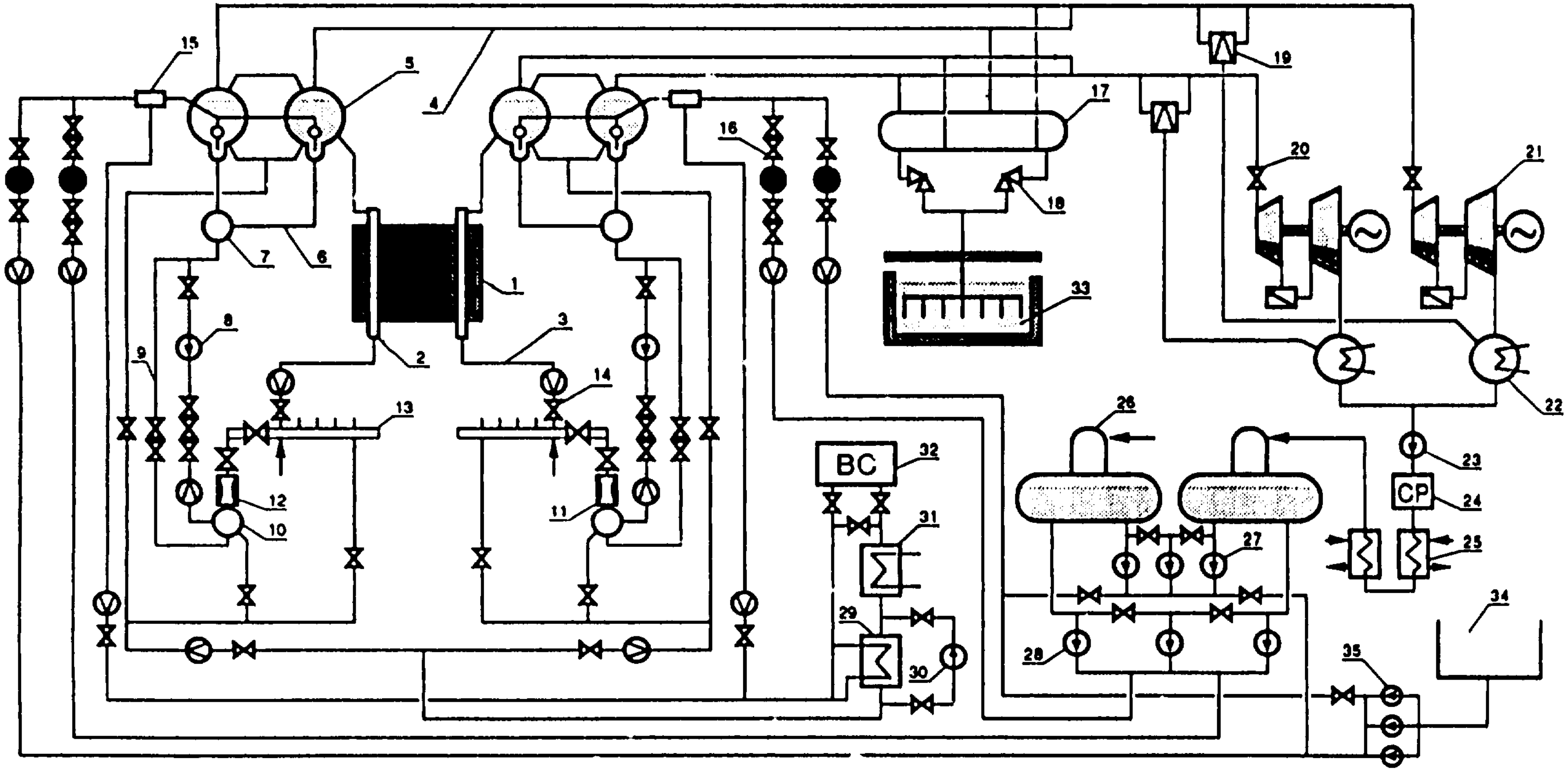
$S = e \cdot L$ = fraction of CS and I inventory leaked to the atmosphere

Table 2 Chernobyl-4 compared to Barsebäck-1
Inventories of a few radionuclides in the fuel

Chernobyl-4 Radionuclide	Inventory MCi	Released fraction of inventory*
85 Kr	0,9	1,0
133 Xe	170	1,0
131 I	86	0,2
134 Cs	5,0	0,1
137 Cs	7,7	0,13

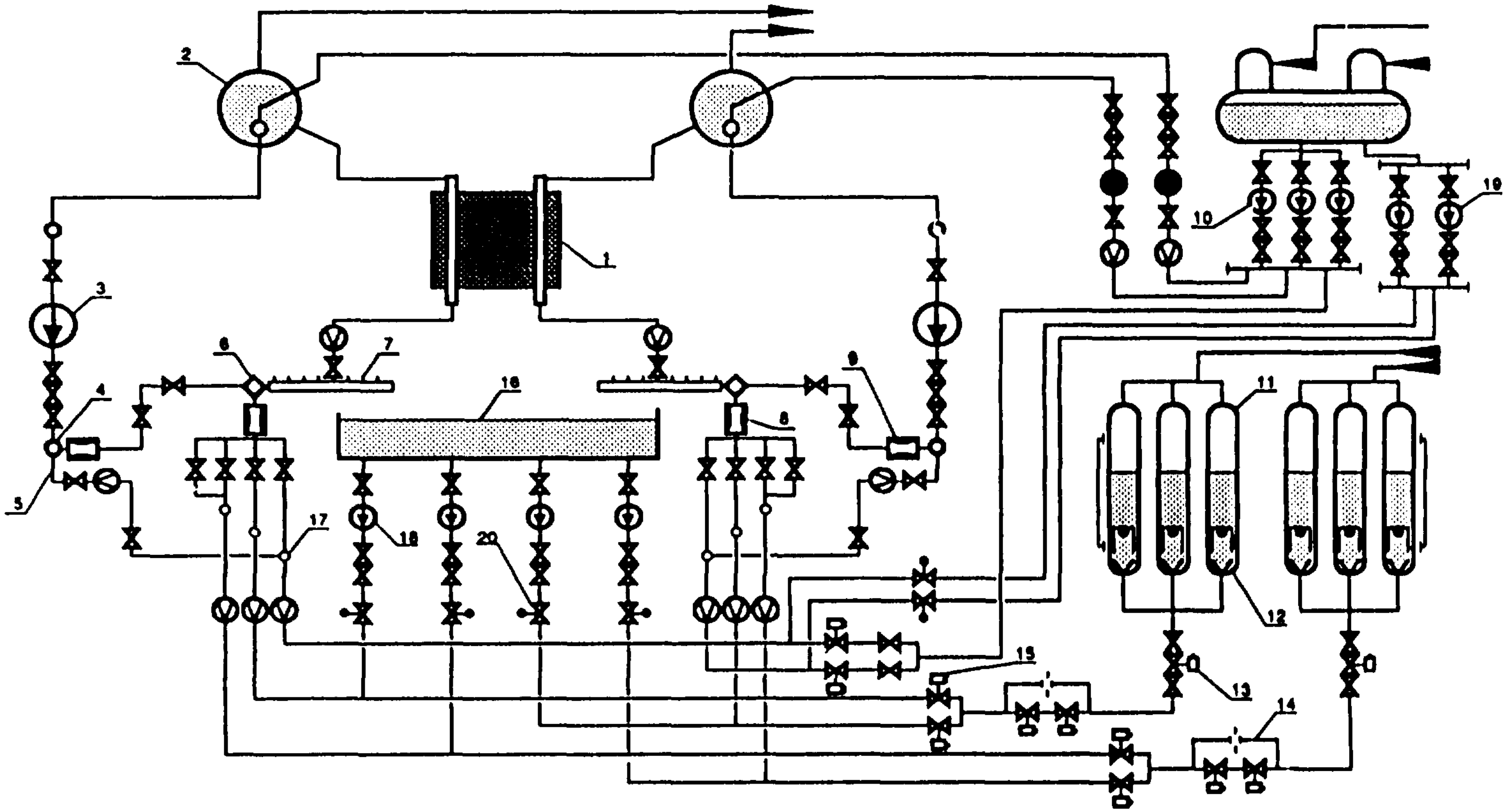
*Referred to 1986 may 6 [14].

Barsebäck-1 Radionuclide	Inventory MCi
85 Kr	0,3
133 Xe	92
131 I	50
134 Cs	4,3
137 Cs	2,7



1 - reactor
2 - pressure tube
3 - water pipelines
4 - steam pipelines
5 - drum separator
6 - downcomer
7 - MCP suction header
8 - main circulation pump
9 - MCP header bypass
10 - MCP pressure header
11 - mechanical filter
12 - flow restrictor
13 - group distribution header
14 - isolation and control valve
15 - mixer
16 - feedwater valve assembly
17 - steam headers
18 - main relief valve (MRV)
19 - BRU-K valve
20 - turbine trip valve
21 - turbogenerator
22 - condenser
23 - condensate pump
24 - condensate purification
25 - heater
26 - deaerator
27 - auxillary feedwater pump
28 - feedwater pump
29 - blowdown regenerator
30 - cooldown pump
31 - blowdown afterheat
32 - bypass purification
33 - ALS pool
34 - emergency water tank
35 - DS emergency feedwater pump

Figure 1
Main flow diagram of the RBMK-1500



1 - reactor
2 - drum separator
3 - MCP
4 - MCP discharge header
5 - ECCS jumper
6 - GDH mixer
7 - GDH
8 - ECCS flow restrictor
9 - GDH flow restrictor
10 - feedwater pump
11 - ECCS pressurized tank
12 - cutoff floating valve
13 - cutoff fast-acting gate valve
14 - intermediate throttling unit
15 - ECCS fast-acting gate valve
16 - localization system condenser
17 - ECCS header
18 - ECCS pump
19 - Auxilary feedwater pump
20 - gate valve of the ECCS long-term pump subsystem

Figure 2
Flow diagram of the Emergency Cooling System of RBMK-1500

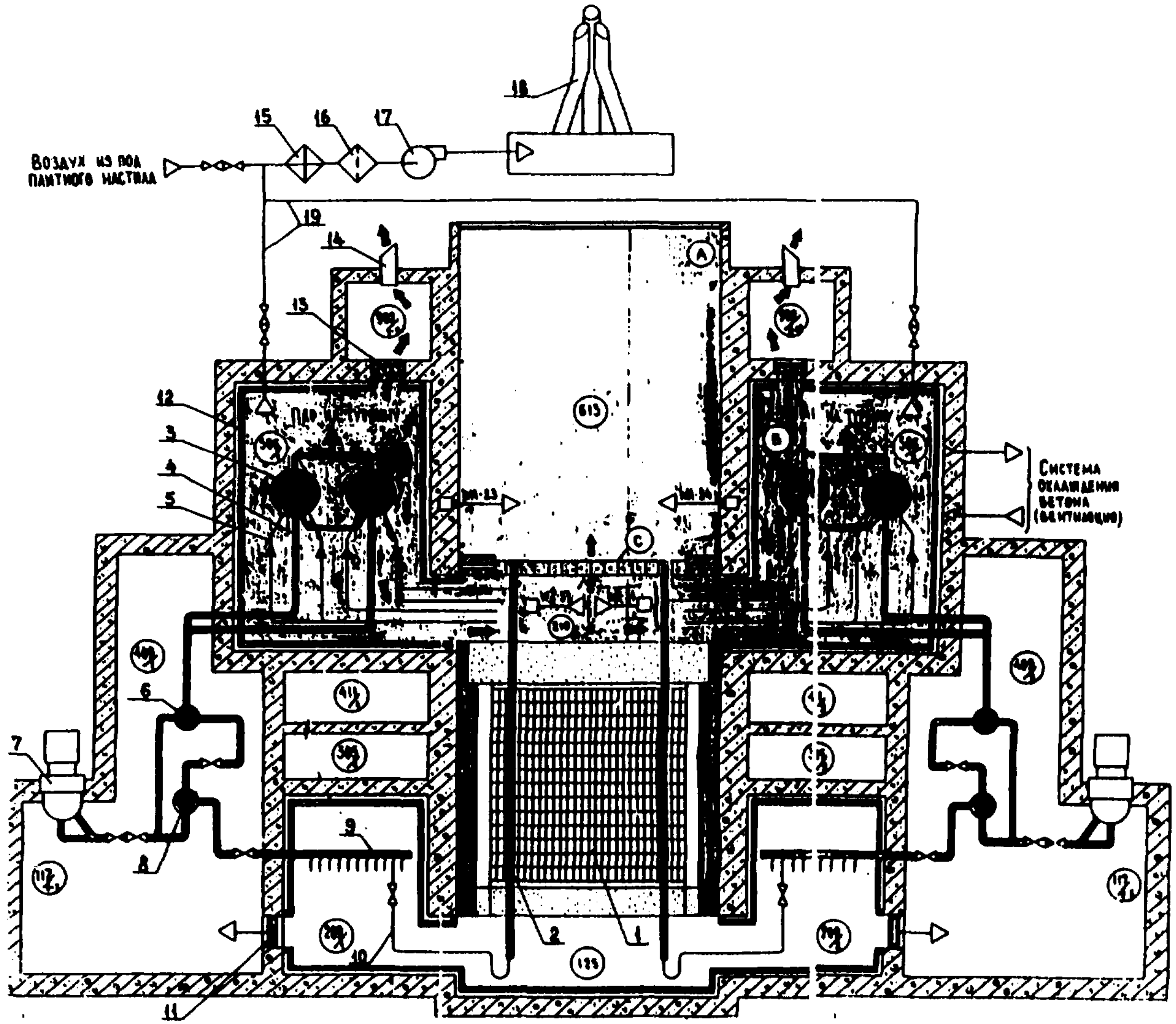


Figure 3
IGNALINA Reactor building, compartments including ventilation schematic

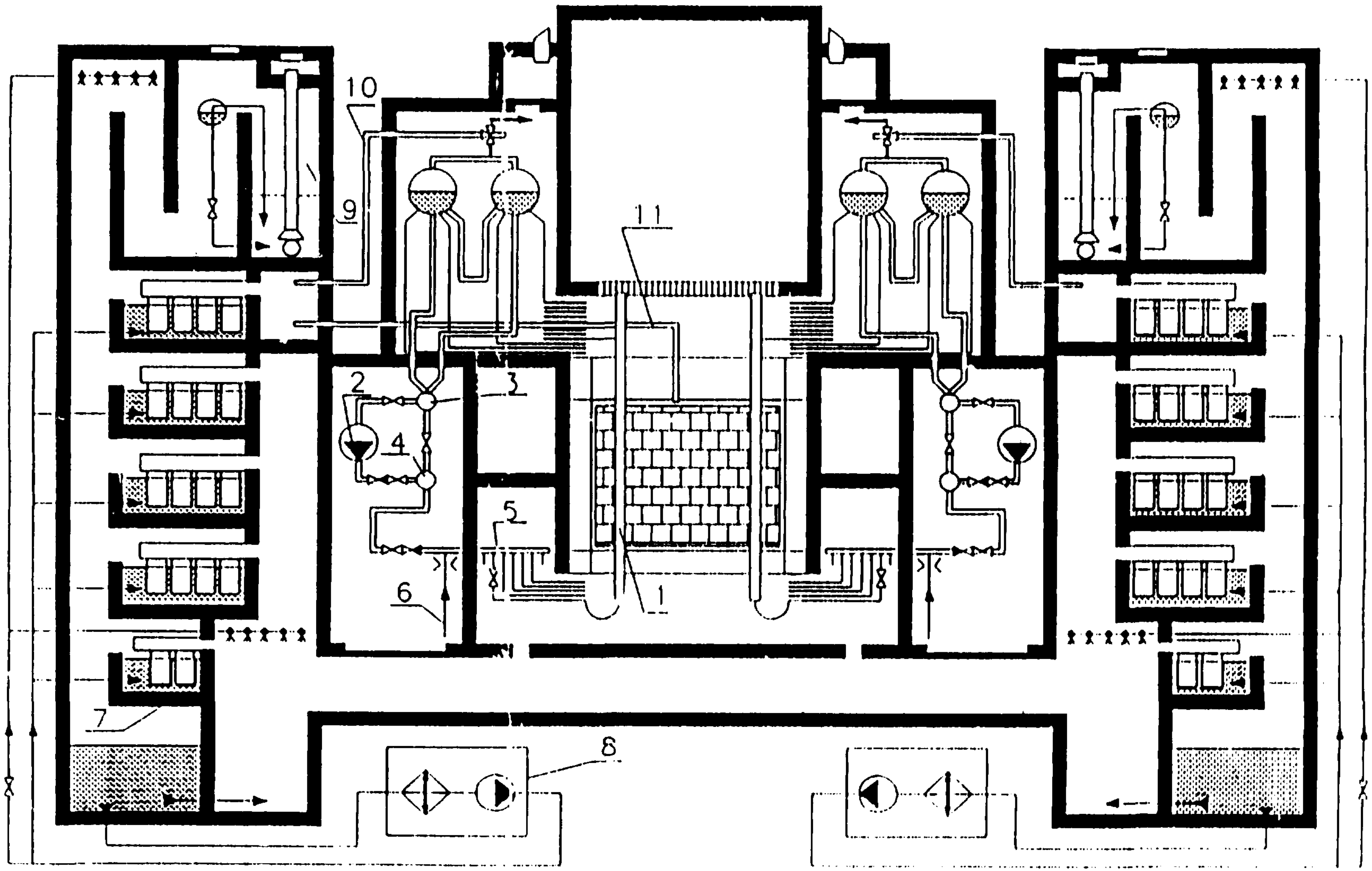


Figure 4
Schematic drawing of the Accident Localization System

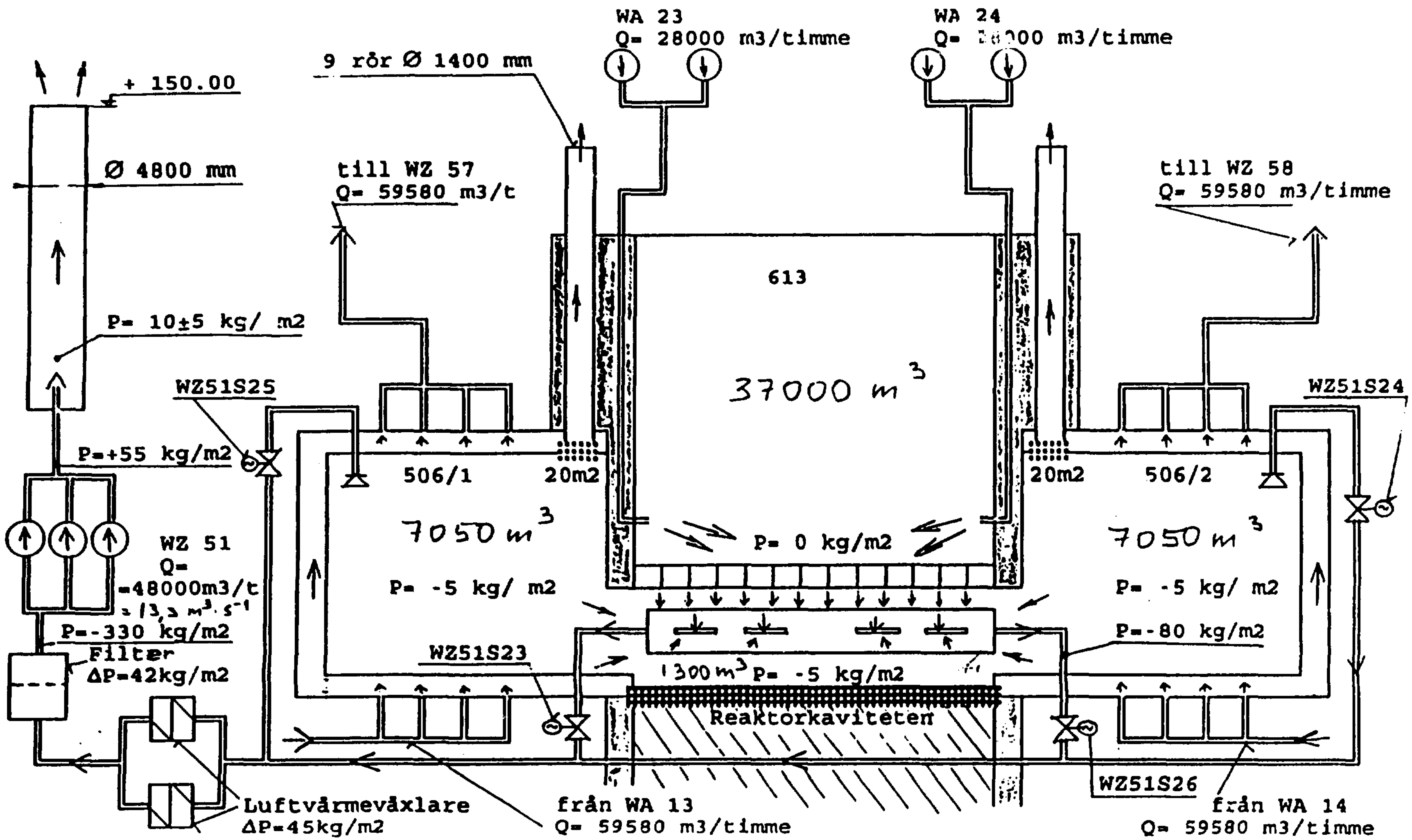
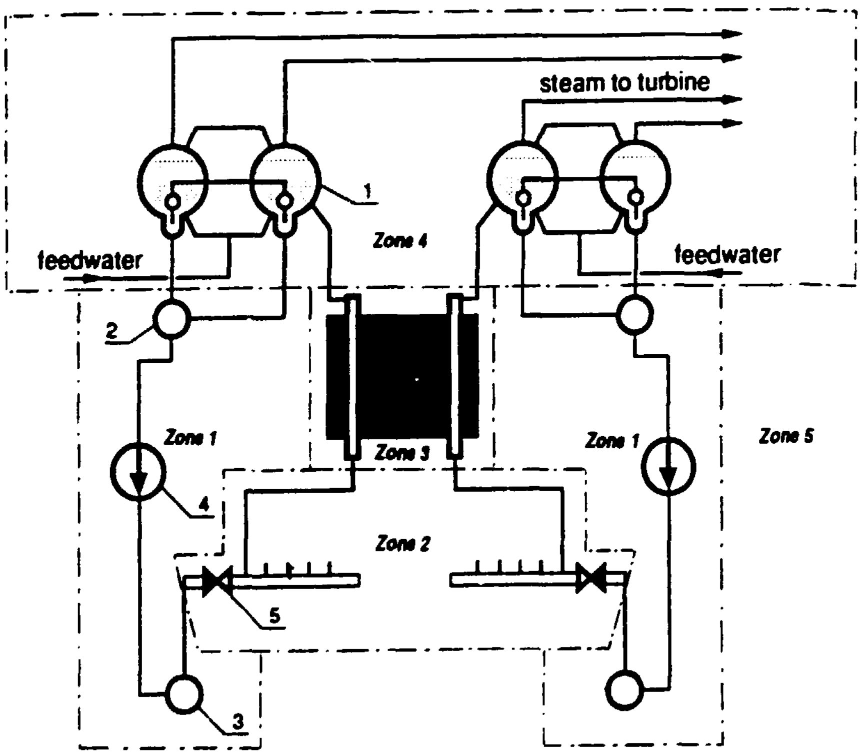


Figure 5
The ventilation system in Ignalina RBMK-1500

Rupture position:	
Zone 1	Inside the primary circuit confinement before (upstream of) the Group Distribution Header (GDH) check valve.
Zone 2	Inside the primary circuit confinement after (downstream of) the GDH check valve.
Zone 3	Inside the reactor cavity.
Zone 4	Outside the primary circuit confinement, on the primary circuit side of isolation chech valves.
Zone 5	Outside the primary circuit confinement, on the secondary side or interfacing system side of isolation check valves.



1 - Drum separator 2 - Suction header
3 - Discharge header 4 - Main circulation Pump
5 - Check valve

Figure 6
Confinement zones in Ignalina RBMK-1500

STUDSVIK/NS-93/47

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Project IBBA, stage 2. Basis for discussion of mitigation measures
founded on natural convection principles

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