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The Flooding Incident at the Ågesta Pressurized Heavy Water Nuclear Power Plant

C. Dahlgren

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C. Dahlgren

Division of Nuclear Power Safety, Royal Institute of Technology,
100 44 Stockholm, Sweden

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SAMMANFATTNING

Den här rapporten är ett examensarbete i Reaktorteknologi utfört på institutionen för kärnkraftsäkerhet, KTH, Stockholm, för SKis räkning.Den är en oberoende utredning utav konsekvenserna av incidenten den 1a Maj 1969, då stora delar av Ågesta kraftvärmeverk översvämmades av vatten. Händelsen aktualiserades efter en artikel i Dagens Nyheter den 23 April 1993, vilken beskrev incidenten och spekulerade i dess konsekvenser.

I denna rapport är utgångspunkten den händelsen, dåen kortslutning inträffade på grund av översvämningen, och ledde till attnöd kylningssystemet (härdsprinklings-systemet) momentant utsattes för ett tryck större än dimensionstrycket. Det hypotetiska scenariot som kommer att analyseras är att ett tungvatten-läckage uppstår i nödkylningssystemet vilket medför att trycket och vattennivån i reaktortanken sjunker.

Rapporten är indelad i tre delar. Den första delen innehåller en genomgång av Ågesta kraftvärmeverk, där de olika systemens funktion och dimensioner beskrivs, en genomgång av incidenten den 1:a Maj 1969 samt utvecklandet av ett hypotetiskt scenario. Det senare förutsätter att ett brott i härdsprinklings-systemet verkligen skulle ha inträffat som en följd av tryckökningen.

Den andra delen innehåller en analys av förloppet efter brottet i det hypotetiska scenariot. Den innefattar en enkel modell av tryck- och nivåsänkningen i reaktorn efter brottet och en noggrann genomgång av flödesgeometrin för läckaget. Dessutom utföres en analys av avkokning av vattnet i primärkretsen, uppvärmningen av bränslet och förstörelsen av härden på grund av otillfredställande kylning av härden i kombination med zirkoniumoxidation. En genomgång av de system som kunde ha använts för att fylla primärsidan med vatten utföres.

Resultatet av förloppet efter brottet blev att nivå- och trycksänkningen hade ett mycket starkt beroende av läckagets volymetriska ång-andel. Denna parameter är mycket svår att beräkna och har i den enkla modellen som utarbetats antagits variera på olika vis. Dessutom har restriktioner vad gäller kritiskt utflöde inte tillämpats på denna modell. Analysen av uppvärmningen och avkokningen i reaktorn gav resultatet att den relativa vattennivån i reaktorn måste sjunka ned till 0.1 (där 1.0 representerar toppen

av härden, 0.0 härdbotten) innan en anmärkningsvärd temperaturökning sker i övre delen av härden. Avkokningen är en långsam process som kan ta flera dygn, vilket leder till slutsatsen att zirkoniumoxidation troligtvis aldrig kommer att inträffa, förutsatt att verket hade byggts om så att lättvatten kan bringas in i reaktortanken.

För att få en mer pålitlig beräkning av initialskedet av förloppet efter brottet på härdsprinklings-systemet utvecklades en komplett modell av Ågesta kraftvärmeverk för koden RELAP5/MOD3.1. Denna modellering utgör den tredje delen i denna rapport.

Transientberäkningarna med RELAP5/MOD3.1 delades upp i ett basfall och flera alternativa transienter. Massflödet i basfallet befanns vara begränsat av kritiskt tvåfas flöde i ca 1 timme (4000 s) efter brottet. Dessutom befanns de dynamiska effekterna, såsom ångbildning i ånggeneratorn, vara av stor betydelse för tryckminskningen och massflödet ut ur reaktortanken.

Efter det att flödet ut ur reaktorn inte längre begränsas av kritiskt flöde och bara består av ånga, kan modellen som utarbetades i den andra delen av rapporten användas för att uppskatta den fortsatta tryckminskningen och massflödet ut ur reaktorn. Termodynamisk jämvikt mellan inneslutning och reaktortank sker enligt denna beräkning efter 4.5 timme (16,000 s). Då har den relativa vattennivån i reaktortanken sjunkit till 0.73. Nivåsänkningen till 0.1 tar ytterligare 4 – 5 dagar, beroende på resteffektsnivån.

ABSTRACT

This report is a Master's thesis in Nuclear Reactor Engineering to be submitted to the SKi (Swedish Nuclear Power Inspectorate). The work has been performed at the Division of Nuclear Power Safety, KTH, Stockholm. It is an independent investigation of the consequences of the flooding incident at the Ågesta HPWR, in Stockholm, Sweden, which occurred on the 1st of May 1969. The issue was raised by an article in 'Dagens Nyheter' on April 23, 1993,which described the incident and speculated about its possible consequences.

The basis for this report is an incident, in which, due to short circuits in the wiring because of the flooding water, the ECCS is momentarily subjected to a pressure much higher than it was designed for. The hypothetical scenario that will be analyzed in this report is the case in which the ECCS subsequently breaks due to the high pressure. As a consequence of the break, the pressure and the water level in the reactor vessel decrease.

The report has been subdivided into three parts. The first part describes the function and dimensions of the different operating systems in the Ågesta HPWR, as well as a chronology of the incident. A hypothetical analysis, based on the event of a break in the ECCS due to the pressure increase, is developed.

The second part of this investigation is a scoping analysis of the incident, whichincludes an energy equilibrium model of the pressurized vessel. It models the pressure decrease and the mass flow rate out of the break. A model of the boil-off in the reactor was also included. The heat-up of the core, and the core degradation was modeled as well. The systems that could have been used to bring water into the primary system during the transients were examined.

The results indicate that the progression of pressure and level decrease after the break is strongly dependent on the volumetric void fraction assumed for the leakage. This parameter is very difficult to calculate, and in the simple thermohydraulic model used in this analysis, no such calculation is performed. The volumetric void fraction of the mass flow of the break is assumed to vary in different ways. Furthermore, other limitations on the mass flow rate in the break flow geometry, such as critical flow, are

not taken into account in the model. The analysis of the heat-up of the fuel and the boil off of the water in the reactor vessel led to the conclusion that a fractional water level of 0.1 (where 1.0 is the top of the core and 0.0 the bottom) has to be reached until significant temperature rise at the top of the core occurs. The boil-off has been found to be a very slow process, which takes several days. The conclusion of the analyses is that zirconium oxidation would never occur, provided that plant arrangements had been made to connect the system for emptying and filling the primary side of its water to an external light water source.

In trying to establish a more accurate picture of the events immediately after the break of the ECCS, a complete model of Ågesta HPWR was made for the RELAP5/MOD3.1 computer code. Of particular interest is the amount of water that is going out from the primary circuit. This model development and the calculations made with the model form the third part in this report.

The RELAP5/MOD3.1 calculations of the transient were divided into the base case and several alternative transients. The mass flow rate was found to be restricted by two-phase critical flow up to about 1 hour (4000 seconds) after the break. Dynamic effects, such as steam formation in the primary side piping of the steam generator, also had a large impact on the pressure decrease and the mass flow rate of the break.

When the mass flow rate in the break flow geometry is no longer restricted by the critical flow, and consists entirely of steam, the model in the second part of the report can be used to describe the remaining part of the transient. Equilibrium between the containment and the reactor vessel is assumed to occur about 4.5 hours (16,000 seconds) after the break. The fractional water level in the core at that point is 0.73. The level decrease down to the point of significant temperature rise in the top of the core takes another 4 – 5 days, depending on the assumed level of decay heat.

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Nomenclature

Arabic:

A	Area [m^2]
C	Constant [-]
C_v	Volumetric heat capacity [J/K]
D	Diameter [m]
E	Internal energy [J]
H	Enthalpy [J]
K	Loss factor [-]
$Level$	Fractional water level in reactor vessel [-]
Nu	Nusselt number [-]
P	Power [W]
Pr	Prandtl number [-]
Q	Heat [J]
Re	Reynold number [-]
T	Temperature [K]
V	Volume [m^3]
W	Work done by the system [J], Formation of hydrogen per area [kg/m^2]
X	Oxidation fraction [-]

c_v	Specific volumetric heat capacity [J/kgK]
e	Specific internal energy [J/kg]
f	Friction factor [-]
i	Iteration step
h	Specific enthalpy [J/kg]
h_c	Heat transfer coefficient [W/m^{2}K]
h_f	Latent heat of fusion [J/kg]
k	Thermal conductivity [W/mK]
l	Length [m]
m	Mass [kg]

$\dot{m}$ Mass flow rate [kg/s]
n Time step
t Time [s]
u Specific internal energy [J/kg]
v Velocity [m/s], Specific volume [m^3/kg]
x Mass fraction of steam [-]

Greek symbols:

Δ Difference
α Volumetric steam fraction [-]
ρ Density [kg/m^3]
ν Viscosity [m^2/s]

Subscripts:

H_2	Hydrogen properties
$STARV$	Starvation constant
UO_2	Uranium oxide properties
Zr	Zirconium properties
ZrO_2	Zirconium oxide properties
bo	Boil-off
$break$	Break properties
$core$	Core properties
d	Decay
h	Hydraulic
i	Beginning of iteration step i
$i+1$	End of iteration step i
$leak$	Leakage properties
n	Beginning of time step n
$n+1$	End of time step n
o	Initial properties
ref	Reference properties
stm	Steam properties
wtr	Water properties

ACKNOWLEDGEMENTS

This report is to be submitted to the SKi and was made at the Division of Nuclear Safety at KTH, Stockholm, Sweden. The purpose is to evaluate the potential consequences of the flooding incident at the Ågesta PHWR on the first of May 1969.

Attention to this subject was drawn by an article in DN in April 1993 [11], which described this incident and the potential consequences that it could have caused. SKi asked the Division of Nuclear Power Safety at KTH, Stockholm, to conduct an examination of this incident.

I want to thank the staff at the Division of Nuclear Power Safety, and especially Mr. T. Okkonen, whose resources in terms of patience and knowledge have been of decisive importance for this report. Also, I want to thank Prof. B. R. Sehgal for providing me with a project that gave me a wide range of experience, from practical engineering work to theoretical analysis.

I also want to thank Mr. S-O. Andersson and Å. Bergman for their helpfuless and patience when helping me find information in the archives of Vattenfall.

Chapter 1

INTRODUCTION

The Swedish nuclear program began in the late 1940's. The Ågesta Heavy Pressurized Water reactor was the starting point for water reactors designed and built in Sweden. It gave essential construction experience and was used as a training facility for operational staff for other Swedish nuclear power plants. The Ågesta nuclear power plant was a pressurized heavy water power reactor (*PHWR*) located in Farsta, approximately 17 km south of Stockholm.

The Ågesta nuclear power plant was one of the first nuclear district heating plant in the world. It was planned and constructed between the years 1956–1964 and it first became critical on the 17th of July 1963. It was then successfully in use for ten years, producing electricity (10 MW) and distributing district heating (55 MW) to the inhabitants of Farsta. In the beginning of 1970, an upgrade of the core power to 80 MW was made. Except for the removal of the fuel and one of the four steam generators, the plant is still in place and is being visited by student groups and others.

The Ågesta reactor was a unique design that is hardly used anymore; the only plant with a similar design concept can be found in Argentina. The concept was that of a large pressurized vessel using heavy water both as coolant and as moderator. The reactor and the primary system were situated within a rock cavity, see Figure 1.1. Ågesta was fueled with natural uranium dioxide (UO_2). The fuel was cooled by heavy water, which was circulated through the core (heat-up) and the steam generators (cool-down). The heat transferred to the secondary circuit containing light water could be used either directly for district heating or for generation of electricity in the steam turbine [2].

The incident described in this report occurred on the 1st of May 1969; it was a typical flooding scenario. It was also an example of the objective reality that an incident in one part of a plant might have severe, unexpected consequences in other parts of the plant.

Attention to this subject was drawn by an article in 'Dagens Nyheter' in April 1993 [11], which described this incident and the potential consequences it could have caused. SKi asked the Division of Nuclear Power Safety at KTH, Stockholm, to conduct an examination of this incident.

The report has been divided into three parts as described below;

- Technical survey of the plant, and an overview of the incident (Chapter 2 and 3). The different parts (and their function and properties) of the plant were described. A chronology of the incident was made, and a hypothetical scenario was considered.
- Scoping analysis of the hypothetical scenario (Chapter 4). A simple thermodynamic model of the leakage was developed, and the outflow geometry was defined. Alternative cooling systems were examined.
- Dynamical two phase flow analysis (Chapter 5). A RELAP5/MOD3 model of Ågesta was developed, and a steady state run was performed. Different transient were run.

It has to be stressed that the hypothetical scenario analyzed in this report never occurred, and that no primary side water leaked out into the containment.

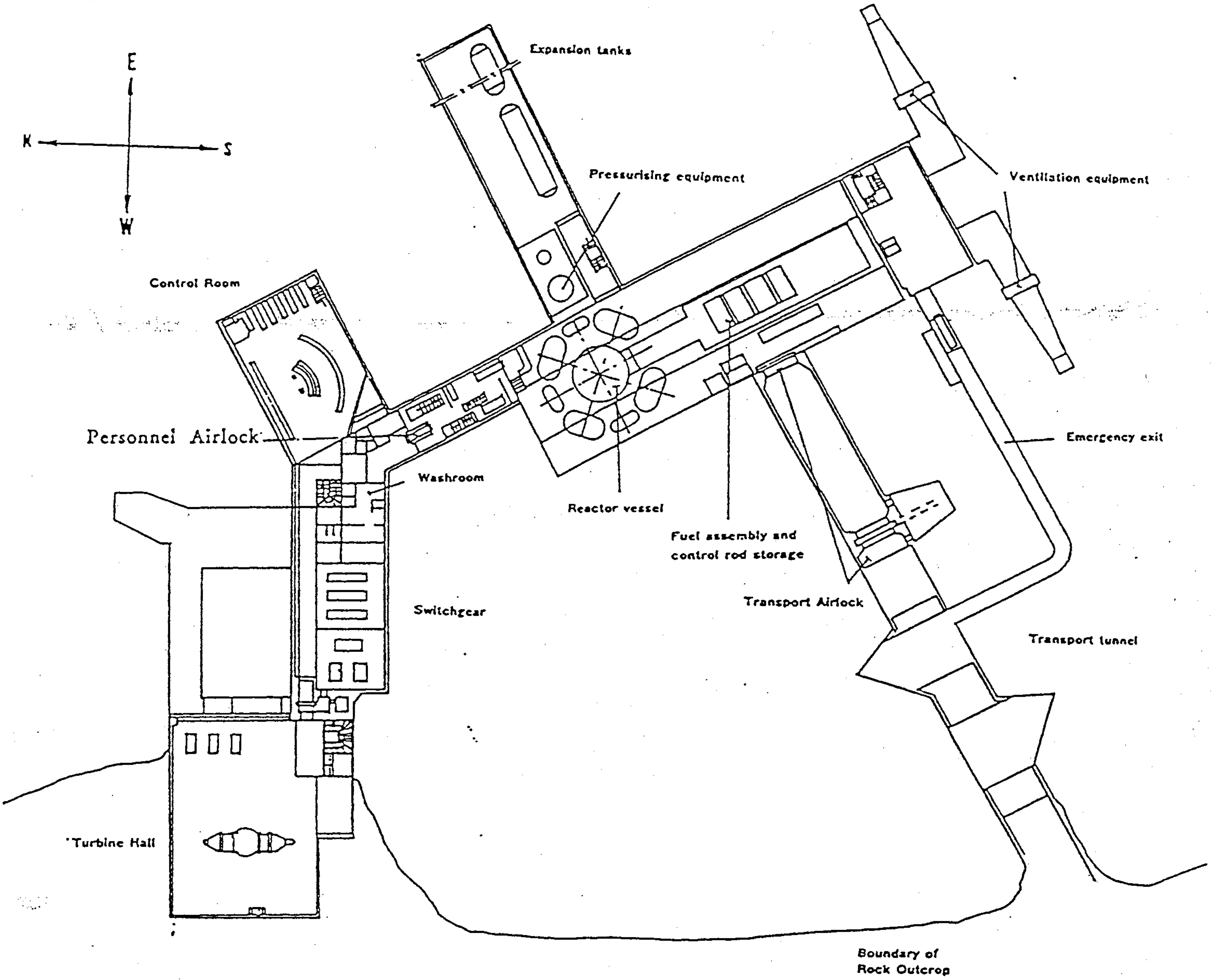


Figure 1.1: Plant Layout

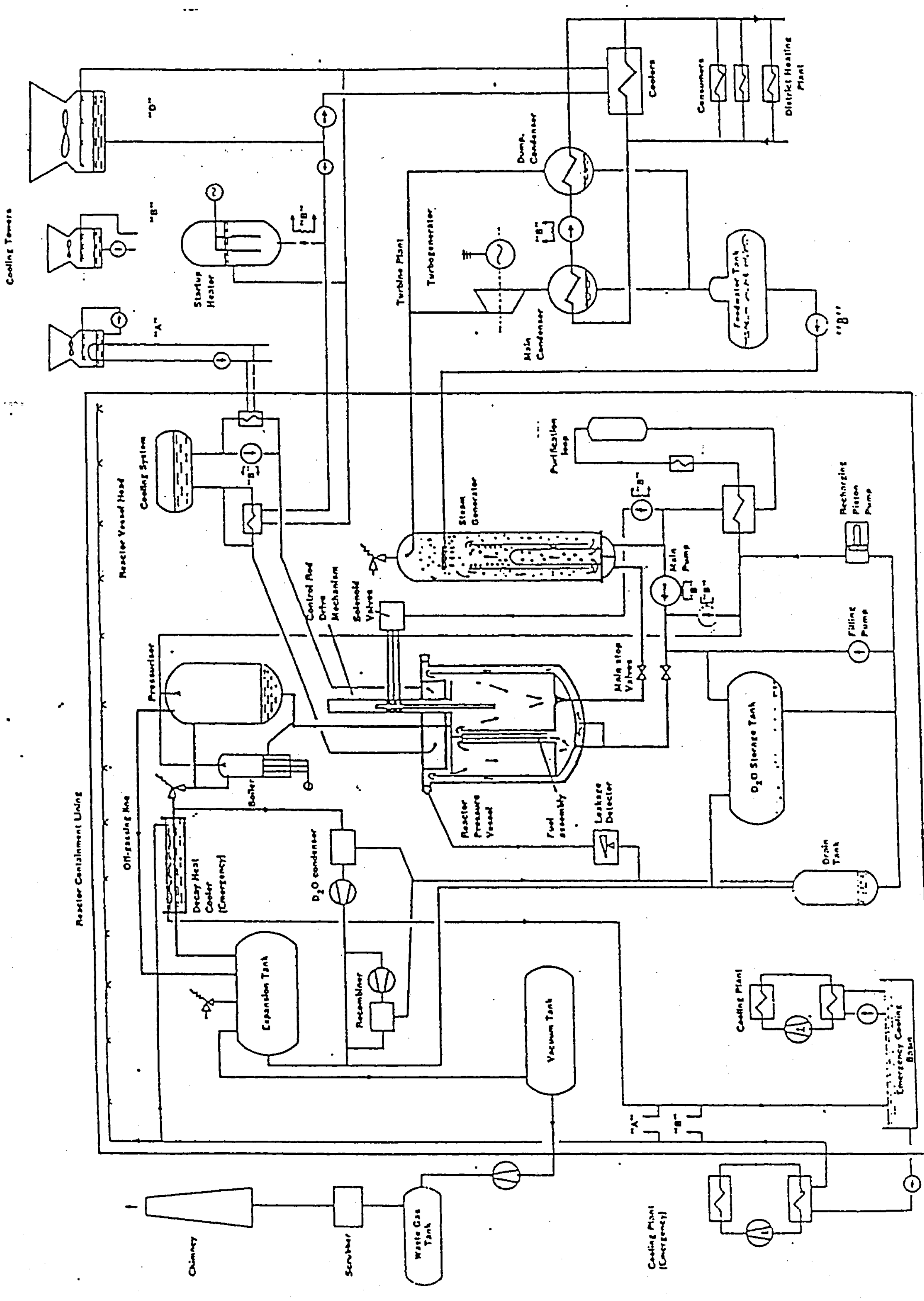


Figure 1.2: Overall Flow Diagram

Chapter 2

PLANT DESCRIPTION

2.1 Location of the Ågesta PHWR

The Ågesta pressurized heavy water reactor is located in Ågesta, 3.5 km south of Farsta, which is a suburb 17 km south of Stockholm. The reactor and the primary system (main pumps and steam generators) are located inside a man-made cavern in a rock cavity. It was placed there for two reasons, namely for protection of the surrounding environment from possible radioactive release in case of an accident and for protection of the plant from possible external attacks [2]. The turbine hall, the cooling towers, the laboratory building, the purification system and the administration building were situated on open ground.

Within a radius of one kilometer from the rock cavern, no buildings were allowed, other than those vital for plant operation. Within a radius of two kilometers from the plant, however, a small population was allowed. In 1964, the population was estimated to be 220 [1, page 7].

2.2 Containment

Most of the equipment of the primary system was situated inside the containment. Inside the rock cavern but outside the containment were the control room, the heavy water support system, the distribution plant and the battery backup [1, page 10].

The containment pressure boundary consisted of 4 mm thick steel walls and roof as well as a 8 mm thick steel floor. It was almost completely surrounded by rock and concrete. Where concrete or rock did not exist, 22 mm thick steel was used. The total volume of the air-tight containment compartment was 29.000 m^3. In order to operate the plant, the containment was penetrated with access holes, system penetrations (piping, electrical wiring and ventilation) and a large ($4 \times 4 \times 24$ m) air-lock. All the penetrations were designed to withstand a pressure of 4 bar [1, page 11]. In case of a large LOCA (Loss Of Coolant Accident), the pressure inside the containment was estimated to rise to 2.5 bar [3]. When the pre-operational testing was performed, the relative integral leakage was measured as 1.55×10^{-4}/h (or 0.372%/day) for a pressure of 2.5 bar [1, page 14].

2.3 Reactor Core and Vessel

The reactor core was situated inside the reactor vessel. The core contained fuel of natural uranium dioxide (UO_2), and the moderator/coolant that was heavy water (D_2O). The core consisted of up to 140 fuel elements, each containing 19 fuel rods. The fuel rods were placed in a modified hexagonal pattern; see Figure 2.1. Each fuel rod consisted of four parts, each part being a tube of Zircaloy (*Zr-2*) that contained 32 pellets of UO_2 and was filled with helium gas. Each fuel rod was approximately 18 mm in outer diameter and 3074 mm in length. The fuel element had a total length of 3750 mm and a diameter of 110 mm [1, page 20].

The design power of the core was 65 MW, of which 60 MW were deposited in the fuel and 5 MW in the moderator [1, page 19]. The height of the core was 3.04 m and the effective diameter was 3.61 m. The mean linear heat rating was relatively low, i.e. 78.5 W/cm. The coolant (moderator) to fuel volumetric ratio was 16.4 [1, page 20]. The core could be characterized as a self-stabilizing core (negative void-reactivity coefficient).

The central rod, and the spacer that held the fuel rods in place, had channels from which the detection system for fuel failures took its bleed-water. The ECCS water was also injected here; see Figure 2.3 [1, page 22]. At the top and the bottom of the fuel elements, there existed openings (junctions) to the moderator, in the form of penetrations in the fuel assembly wall, to provide more efficient natural circulation.

To control and shut down the nuclear reaction, 29 control elements with neutron absorption by an alloy of 80% silver, 15% indium and 5% cadmium were placed throughout the core. The control elements were all canned in stainless steel. Two of them were controlled with fine rod drives, whereas the 27 remaining had coarse rod drives [2].

The reactor pressure vessel was a cylinder made of 65 mm thick steel with a ellipsoidal bottom head and a flat lid; see Figure 2.2. It had an outer diameter of 4890 mm. The total volume of the vessel was 50 m^3. The temperature of the lid was kept equal to that of the vessel by directing a light water flow through it. The total height of the vessel was approximately 7 m. The vessel had an internal stainless steel liner that was 5 mm thick. The lid was connected to the vessel by 48 bolts. The tightness of the lid was ensured by a ring of silver, which was plastically deformed when the lid was connected to the tank, and a toroidal seal. The total weight of the vessel, without piping and insulation, was 296.6 tons [1, page 34].

Thermal shields were used to protect the steel in the vessel from embrittlement due to neutron bombardment. The upper thermal shield consisted of thick square slabs of stainless steel. The radial shield consisted of two concentric cylinders and the bottom thermal shield of the water flow distributor [1, page 34].

The lid was penetrated by many holes of different sizes, among others 37 loading holes (for control rods and refueling) and 140 pipes for fuel failure detection (and the ECCS injection) [1, page 34]. It was cooled by light water, to keep the temperature equal to that of the vessel. The maximum temperature difference between the vessel and the lid was 10°C [1, page 31].

2.4 Primary System

2.4.1 Main Coolant Circuit

The primary system was a closed circuit containing pressurized heavy water. The heavy water transferred heat from the fuel elements to four steam generators, where primary heat was transferred to the secondary system. The water flow was circulated by four main circulation pumps, one in each loop, located downstream of the steam generator.

Water flows in the vessel through two parallel routes [1, page 34];

- The fuel cooling flow took care of the cooling of the fuel elements. From four inlets in the bottom of the vessel, heavy water was brought into the core and distributed to the 140 fuel elements. The heavy water flowed upwards through the elements and then downwards through the moderator volume. The heavy water left the vessel through the four outlets, also located at the bottom of the vessel.
- The annulus flow went through the annulus between the vessel and the outer thermal shield. The annulus flow then joined the moderator flow. The purpose of the annulus flow was to cool the thermal shields and to cool the vessel itself. The flow kept the temperature of the vessel walls of the vessel equal to that of the fuel inlet cooling flow.

The heat released from the fuel in the pressurized vessel was 65 MW. It was transported by a heavy water flow of 1020 kg/s. The temperature of the heavy water was increased by 15°C, from 205°C to 220°C, during its passage through the vessel [1, page 49]. Under normal conditions, it took approximately 50 s for the water to enter and leave the vessel.

The nuclear process was coupled with the heatup of the coolant/moderator, therefore the moderator could automatically control the nuclear process (if more nuclear fissions take place, the moderator temperature increases and the fast neutrons are not slowed down as efficiently as before so that fewer thermal fissions take place). Because of the large volume of the moderator ($V_{D_2O}/V_{UO_2} = 16.4$), the system was self-regulating and stable. While the inlet temperature could be altered to change the power level (the lower the inlet temperature the higher the power), the outlet temperature was kept constant [1, page 48].

2.4.2 Pressurizer

The heavy water had a high pressure, 34 bar, to prevent boiling [1, page 48]. To control the pressure in the reactor vessel, it was connected to a pressurizer 20 m^3 in volume. The bottom of the pressurizer was located about 5 m above the top of the reactor vessel. The pressurizer was connected to the vessel top by four 225 mm pipes. During normal operation at an outlet temperature of 220°C (which is equal to a subcooling of 20°, at the operation pressure of 34 bar) steam was generated in the electrical boiler. The heavy water volume in the pressurizer vessel was 1 m^3. The pressurizer also contained some D_2 and N_2 gases. The pressurizer served both for pressure control and relief. The design pressure in the system was 40.2 bar. Pipes from the pressurizer went to the expansion tanks, which were also designed to handle the heavy water steam that would be produced in the event of a LOCA and afterwards. During normal operation the expansion tanks were used to collect the leakage of heavy water [2].

2.4.3 Steam Generators

The four steam generators were of the 'inverted tube in shell' type [2]. They were mounted vertically, with the heavy water inside the tubes and the light water on the shell side. At full power, steam was generated at 196°C and 14.3 bar. The steam flow was 30 kg/s. The tubes had an outer diameter of 10 mm and an inner diameter of 7.5 mm. The steam generators had a heat exchange area of 500 m^2 each. The steam generator shell was made of 150 mm thick stainless steel [1, page 44f]. During startup, the steam generators were used for warming the heavy water of the primary side with electrically heated light water on the shell-side. The steam generators were 10.5 m high, and had an inner diameter of 1.7 m. The volume of the secondary side in each steam generators was 22 m^3; see Figure 2.5.

2.4.4 Main Coolant Circulation Pumps

The four main coolant circulation pumps were of the one-step centrifugal type. Each pump had a flow capacity of 255 kg/s, the pumps used 129 kW of power altogether at a pump efficiency of 81%. The pumps had two speed drives.

2.5 Secondary System

The secondary system contained light water, which was used for generating electricity and district heat in the steam plant. The steam plant consisted of the steam turbine, the condenser system and the feed water plant [1, page 62].

- The steam piping brought the steam that was produced in the steam generators to the steam turbine and/or the dump condensers.
- The condensation plant brought the condensed water from both the turbine condenser and the hot-well to the feed water tank. It also brought the condensed water from the dump condensers to the feed water tank.
- The feed water system included pumps pressing the water from the feed water pumps to the steam generators and to the feed water tank. It was also used for bringing the water to the startup heat exchanger when warming up the feed water tank. Finally, it was used for the storage of feed water during de-gasifying.

Since the system was planned to keep the outlet temperature and the heavy water flow constant through the reactor vessel, the reactor power was controlled by altering the pressure on the light water side in the main heat exchangers. If the turbine load increased, the steam consumption increased while the steam pressure (and the temperature) in the steam generator domes was decreased. Then, the temperature on the primary side dropped — and thereby the core inlet temperature. More power could then be taken out from the reactor.

The plant had two separate ways of operating the secondary side; the control by bypass cooling and the temperature control [1, pages 69–70]. In the temperature control mode, the load on the district heating side decided the reactor power. Temperature control means that the temperature of the outgoing district heating water was kept constant.

2.6 Recooling System

If the reactor power had to be kept at a certain power level regardless of the district heat consumption (for example, when full electrical power from the steam turbine was needed, or for reasons of reactor engineering), the hot water that returned from the district heating grid was cooled by the recooling system, in case of a turbine failure a dump condenser could take the reactor heat load. Connection was made from the turbine hall building to the cooling towers, which could accept the heat in the event of a failure in the district heating system [2].

The recooling system (P210) consists of 4 separate units, each containing the following; a propeller, 4 water spreaders, heat exchanger bodies which are made out of paper with a plastic surface and a water cleaner which separates any particles from the water and cleans the water from algae. All the equipment were located in a cooling tower and the water was kept in a big basin ($500m^3$)

The recooling system was designed to absorb a large amount of heat. It was used for the following purposes [1, page 181]:

- To cool the surplus heat of the district heating grid when the reactor is not operated and a constant temperature is desired.
- Residual heat removal on certain occasions.
- To cool the core during refuelling.

2.7 District Heating System

The district heating system was designed to provide hot water for the 10,000 households in Farsta. The distance from Ågesta power plant to the local, oil-fueled subheating plant was 3.5 km.

The district heating water that was heated in the secondary side condensers of the plant was piped to the central heat exchanger and the warm water pipes in each house. The district heating piping system was a closed circuit. The outlet and inlet temperatures of the district heating water ($T_{out} = 75 - -120°C$, $T_{in} = 50 - -20°C$) varied with the demand. The large amount of water and the long circulation time made it possible to shut down the reactor for short periods of time, without causing any inconvenience for the customers [1, page 78].

2.8 Emergency Core and Containment Cooling System

The Emergency Cooling System of Ågesta reactor was subdivided into two parts [4]:

- The Core Spraying System (denoted ECCS) P214/P82 prevented over-heating of the fuel in case of a LOCA. Light water was sprayed on the core through the fuel failure detection system (P82), which was a system which monitored temperatures and radioactive content of the primary side coolant during normal operation (its characteristics are described in further detail in section 4.4). In a large break LOCA, the core spraying system was vital for preventing core damage.
- The Containment Spray System (denoted CCS) P214 was designed to cool down and depressurize the containment after LOCA. Containment spray lines were installed in the steam generator rooms, the rooms where the expansion tanks of the plant were located, and the reactor hall.

The water that was used in both spray systems was taken from a basin that has a volume of 600 m^3. The basin was insulated with styrofoam and the water was kept cool (about +2°C) by a cooling unit. When an accident had taken place and both spray systems were working, the water from the core spray system and from the containment spray system was collected in the drainage system and kept in a separate basin to cool down [4]. After approximately 45 minutes, the water level in the large cooling basin would be low and the water from the blowdown would then be pumped back there again from the drainage system (mixing heavy and light water). During the first 6 hours of operation, the temperature in the large cooling basin increased from +2°C to +15°C [4].

2.8.1 Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) was designed to give a cooling effect enough to handle the heat released after a large break LOCA (Loss Of Coolant Accident), which in this case was assumed to be a break of one of the main coolant circulation pipes. The containment would then receive the heavy water content of the vessel plus the light water contents of one of the steam generators (when one of the pipes break, the steam generator connected to that pipe was assumed to lose its light water as well, but the other three steam generators are not affected). In such an accident, the core coolant would escape from the reactor system, therefore, the core was sprayed with water, to prevent the fuel from over-heating.

The ECCS consisted of two pumps, P214 B1 and B2, each with the other one as a backup, if the activated pump showed any signs of malfunction, see Figure 2.6. Each pump had a capacity of 33 kg/s at 11.8 bar. The pumps were located outside of the air-tight containment [4, page 4] and [5, page 8], and took their (light) water from the big basin that contained 600 m^3 water, at a temperature of +2°C. The valves for opening and shutting the system were placed upstream and downstream of the pumps. All of them were opened and closed by motors. The ECCS piping was connected with three valves to the system for fuel failure detection [4]. The light water flow through the core spray system into the vessel was 2000 kg/min (33.3 kg/s), it was distributed over the 140 fuel elements. Each fuel element was to receive approximately 15 kg/min of light water. The flow was distributed over the cross-section of the fuel elements [9]. All power to the system was taken from the power lines backed up by the diesel generators.

The ECCS was actuated if any of the following three conditions were fulfilled [6];

- The pressure in the vessel was lower than 7.5 bar.
- The temperature in the steam generator room was over 60°C.
- The water level in the vessel was lower than 0.7 m (from the top).

Because of possible malfunctions when opening the valves that connected the ECCS to the system for fuel failure detection, cross connections were made between the three lines of piping to ensure distribution of water over the whole core [6].

2.8.2 Emergency Containment Cooling System

The Emergency Containment Cooling System (CCS) was designed to lower the over-pressure that would be created in the containment or in the reactor after an incident. It was also designed to keep the containment at a pressure below the atmospheric pressure for a long time, approximately 100 days. The low pressure would prevent any radioactive gases from escaping out from the containment.

The maximum pressure in the containment after a large break LOCA was estimated to be 2.5 bar, and the containment spray system was designed to reduce that to a pressure of 0.995 bar in an hour. The low pressure was then kept for 100 days, a time necessary for decay of the noble gases and the short-lived radioactive iodine that may have accumulated in the containment. The temperature in the containment after a large break LOCA was calculated to be 105°C. That would be reduced to normal indoor temperature, 20°C [6].

The containment spraying system consisted of three pumps, P214 B3 - B5. Each pump had a capacity of 45 kg/s, with a pressure head of 8.3 bar. The containment spraying system took its water from the same basin and used the same piping as the core spraying system. The water that was pumped from the basin passed through three emergency pre-cooling units before it was sprayed. From the emergency pre-cooling units, five pipes led to different parts of the containment; the main steam generator room, the room where the expansion tanks of the plant were located, the reactor hall for coarse spraying, the reactor containment with fine sprinkling (two lines). The valves for these operations were all opened and closed with motors [4].

Parts of the containment spray system could be activated individually, regardless of other parts of the system or the core spray system. That was the case when the pressure difference between the main steam generator room and the environment exceeded 0.025 bar [6].

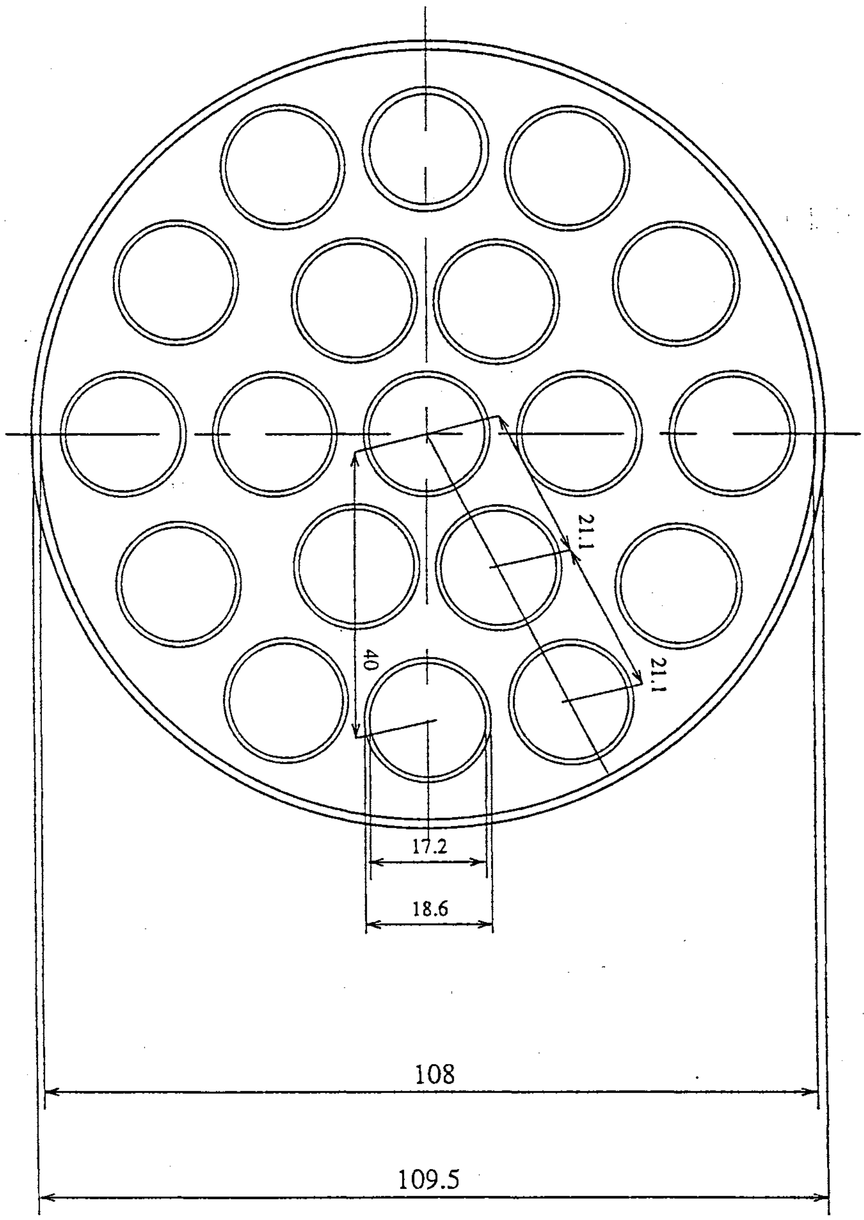
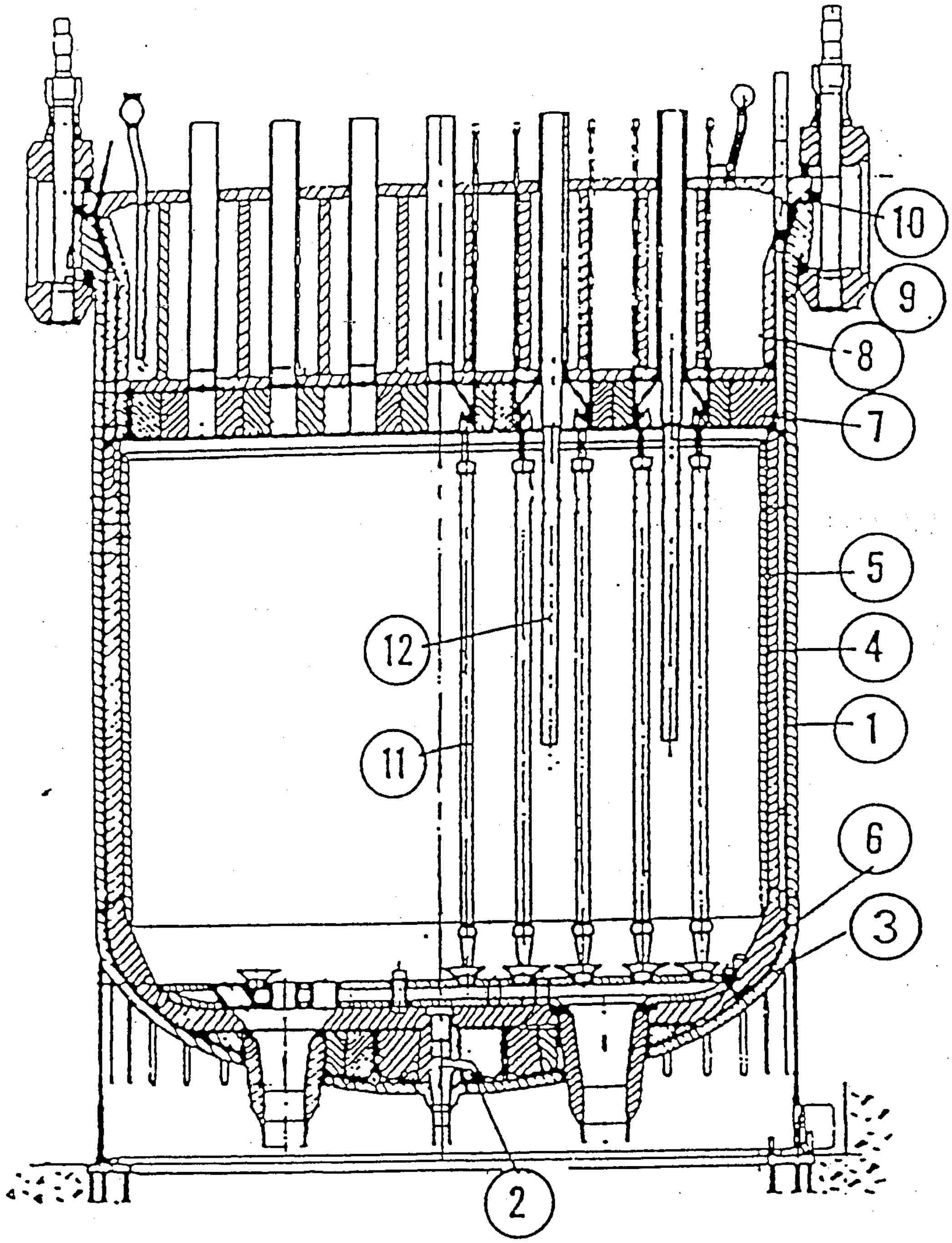


Figure 2.1: Modified Hexagonal Fuel Pattern



1. Vessel wall
2. Filler ring
3. Filler body
4. Outer thermal shield
5. Inner thermal shield
6. Distribution header
7. Upper thermal shield
8. Vessel head
9. Flange
10. Silver seal ring
11. Fuel assembly
12. Control rod

Figure 2.2: Pressurized Reactor Vessel

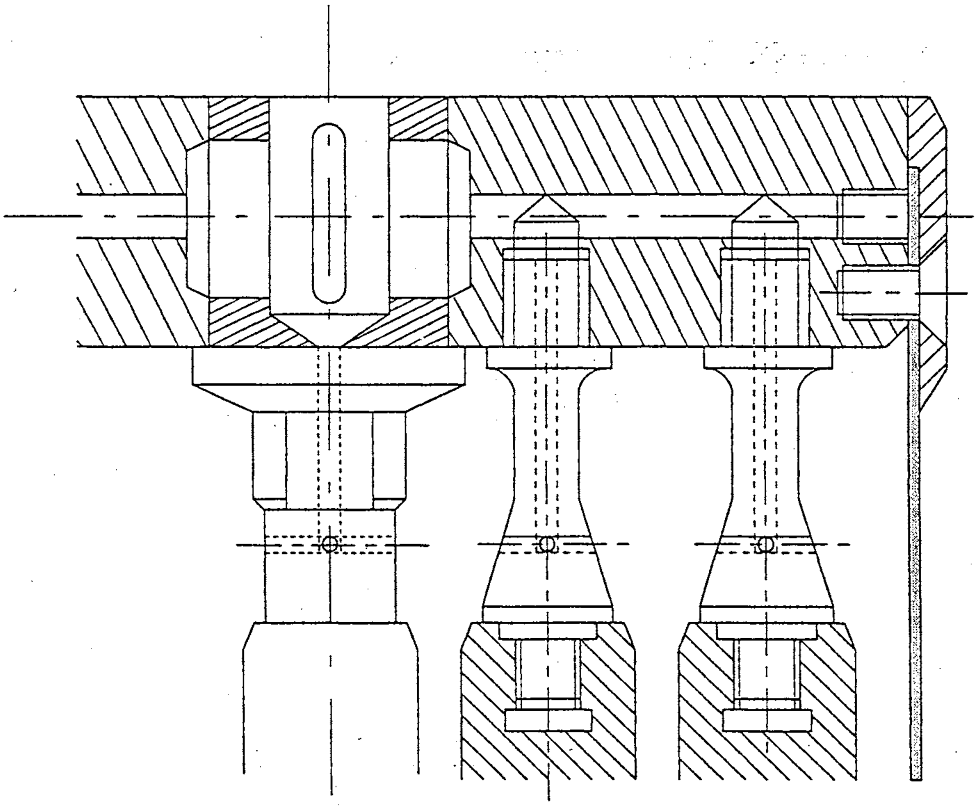


Figure 2.3: ECCS Water Injection Into Fuel Element

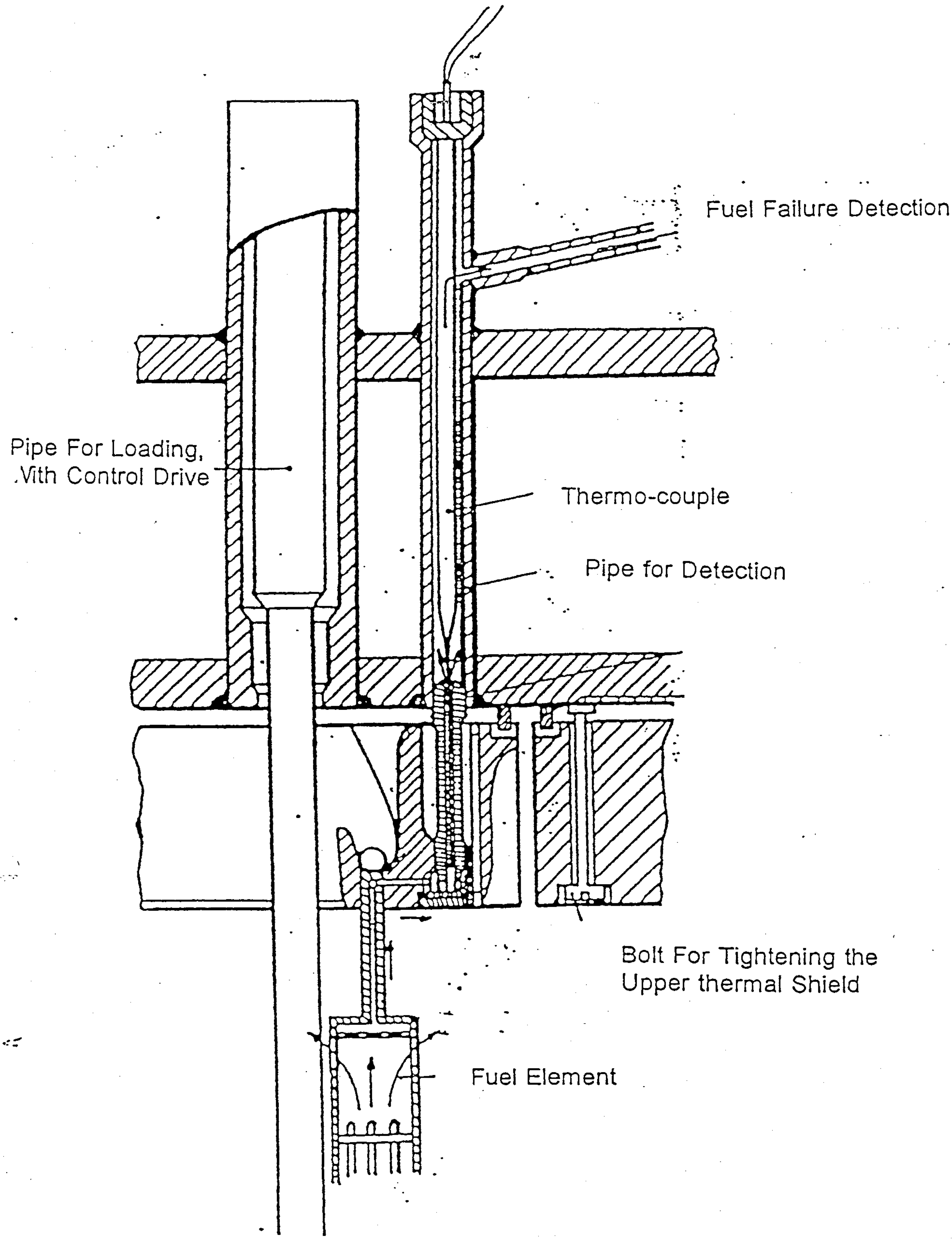


Figure 2.4: ECCS Water Outlet.

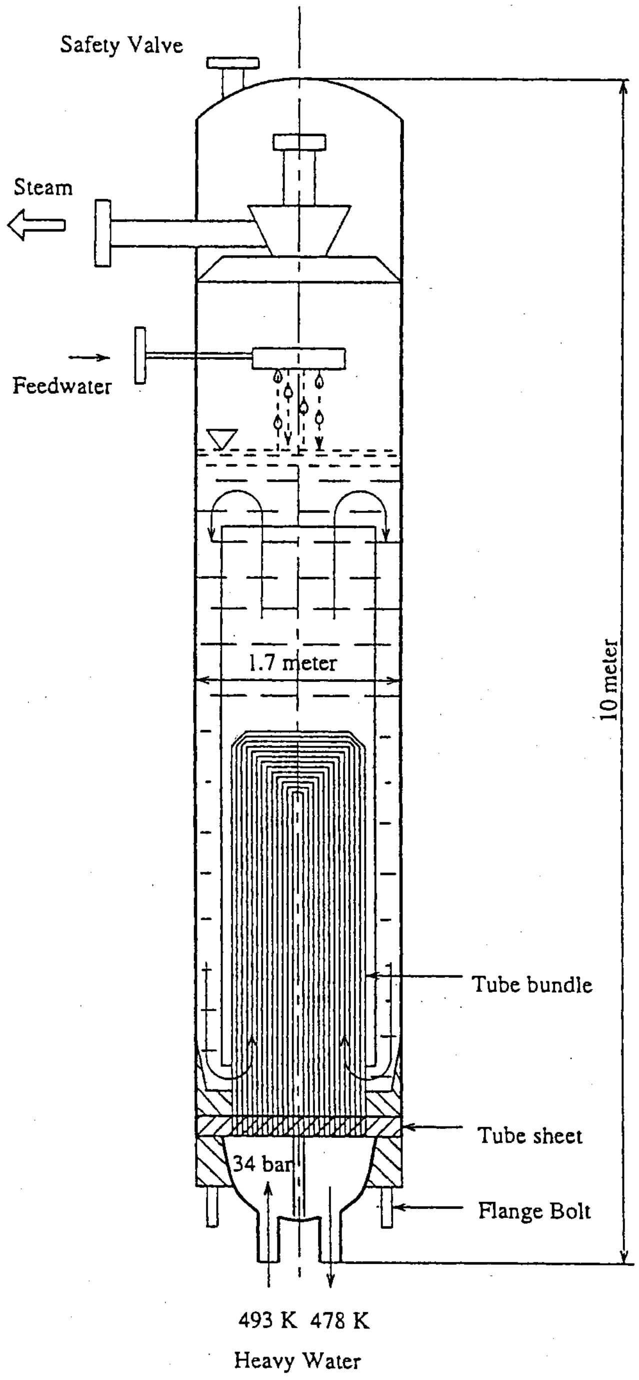


Figure 2.5: Steam Generator

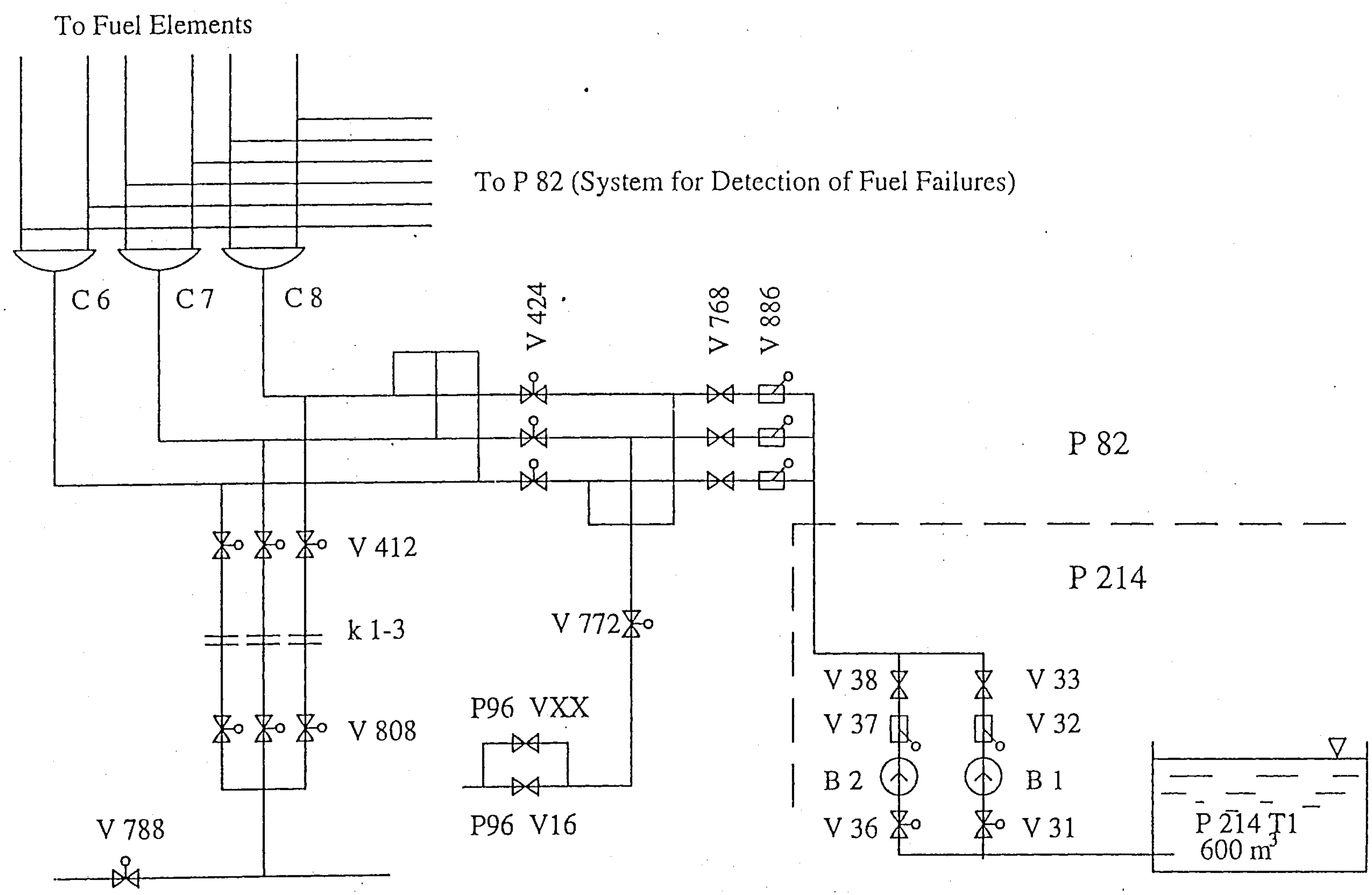


Figure 2.6: Emergency Core Coolant System Flow Chart

Chapter 3

THE FLOODING INCIDENT AT THE ÅGESTA NUCLEAR POWER PLANT

3.1 Review of the Flooding Incident

The incident that took place in Ågesta on 1st of May 1969 was a typical flooding scenario. It was also a good example of the fact that an accident in one part of a nuclear power plant, or any complex industrial plant, may cause severe unexpected consequences in other parts of the plant.

On the 1st of May 1969, the Ågesta HPWR had been producing electricity and district heat with full power (55 MW_{th}, and 10 MW_e) since October the year before. In the morning of that day a switching between two pumps, P210 B1 and B2 which were working in the recooling (P210) system, was performed according to instructions. In this operation, however, valves were opened and closed in an incorrect order [7] and [8]. This resulted in a rupture in one of the valves, which caused a major leakage of cooling water. One of the large basins in the recooling system was emptied of its contents of 400 m^3 of light water. The stream of water hit a main busbar of the generator and caused a turbine shut-down. Due to the short-circuits in the wiring, flooding was not indicated on the control board, even though a ground fault was recorded.

The flood water caused many short-circuits in the electrical wiring and other problems. The water ended up in a room in the turbine building and the drainage system became overloaded. In that room pumps for the system P242 (the emptying of feed water and condensed water) and two drainage wells were located. Subsequently, the water ended up in a separate basin and in the big cooling basin used in the ECCS. These basins were flooded and water had to be pumped out into the Lake Orlången.

In another area of the plant a plug for cleaning the piping of the drainage system (a "cleaning eye") was pressed out. Water then flooded through it. Unfortunately, the plug was located right above a cubicle, where many of the vital relays for the ECCS (P214) were located. When the cubicle was flooded with water, several of these relays were short circuited. The core spraying system was activated, one of the pumps started (P214 B1) and two valves opened, connecting the primary side water with the ECCS. Consequently, the other pump of the system also started automatically.

The ECCS (Emergency Core Coolant System) worked at the pressure of 16 bar and the piping was designed to withstand as much as 20 bar [11]. The reactor was not shut down at the moment when the core spraying system started, which means that the piping in the ECCS was exposed to a pressure that was probably much higher than it was designed for (the reactor pressure was 34 bar). Instead of working as it normally should, pumping in light water into a de-pressurized core (which would have been the case after a large break LOCA), a total of 0.46 m^3 of heavy water was pushed into the piping of the ECCS. One possible reason that the piping did not break would be a leakage in the isolation valves that separated the system for detection of fuel failures (P82) from the ECCS (P214), see Figure 2.6.

Half an hour later, two isolation valves on the primary side closed due to further short circuits. The reactor was then shut down, to prevent the steam generator from being emptied of its contents.

About 8 hours later, the room that was flooded with water could be entered. Because of the serious damage to the electrical components, and the large amount of water in the plant, it was decided that the reactor could not be restarted immediately. It was considered that the shut down time would help to dry out the systems and eliminate the possibilities of new ground faults.

3.2 Chronology of the Flooding Incident

1st of May 1969

09.14 According to instructions, a routine change of a pump is started in the recooling system P210 (used to control the temperature of the incoming water from the district heating grid). The pump in the recooling system was operating one month at the time, the other pump was on standby.

09.15 A valve located downstream of the pump that was recently shut down ruptures. The cooling water pond, which contains the cold water used to cool the incoming water from the district heating grid is emptied of its contents; 400 m^3 of light water. The pond is located at the level +30 m.

A stream of escaping water from the ruptured valve hits the main busbar of the generator (6 kV). This leads to a turbine trip.

No indication of the flooding is given on the control board. Another signal in the control room showed, however, that the turbine shut-down was due to a ground failure.

09.15 – 09.20 The water continues to flow down through the plant, filling a room of the turbine hall building with water, causing several electrical short circuits on its way.

09.20 – 09.25 The drainage system in the turbine hall becomes overloaded. Now the plug for cleaning the draining pipe in another part of the station is pressed out. Unfortunately the plug is located right above a cubicle (called MA13) where many of the vital relays of valves and pumps of the ECCS are situated.

When the basins in the drainage system become full, the water flow is directed to the lake Orlången. (Only a small amount is pumped out. The radioactivity is measured later and no increase is recorded.)

The cubicle is flooded with water. This leads to several short circuits in the wiring, one of which leads to the startup of both ECCS pumps. Also, two of the valves that separate the heavy water of the primary system (the system for detection of fuel failures) and the light water of the ECCS are opened shortly after the pumps are started. The service personnel shut off the ECCS manually, but did not realize that the valves had opened.

When the valves opened, the pressure in the piping of the ECCS momentarily increased from the operating pressure of the ECCS (16 bar) to the pressure in the primary system (34 bar). The piping in the ECCS was only designed to withstand pressures as high as 20 bar, yet it did not fail (because of the leakage in the isolation valves V886 in Figure 2.6).

Instead of pumping water to a depressurized core, 480 kg of heavy water flowed from the primary circuit into the ECCS. If the ECCS piping had broken, heavy water would have been released from the pressure vessel and the core would have suffered a small break accident. If a break would have happened outside the containment, the heavy water would have been released into the environment. It should be emphasized that the reactor had only one emergency cooling system.

09.42 The reactor is shut down, after shutting of two isolation valves on the feed water piping in the secondary system, caused by other short circuits in the wiring.

09.43 – 09.44 Two of the three pumps of the containment spraying system start and the valves of the system open. However, the system is shut down manually after a short time. A small amount of water was sprayed into the containment and the steam generator room.

09.45 and later Because of electrical failures, the valves of the ECCS (P214) opened and closed a few more times during the day, probably causing further pressure peaks in the piping of the core spraying system. A total of 15 main electrical failures are observed, among others problems with indicators being 'half-lit' instead of not shining at all [8]. Other valves are also shut or change function due to short circuits in the wiring.

17.15 The room that was filled with water can be entered, when some of its content has leaked out through the door [11].

5th of May 1969

Nuclear restart of the plant. All the electrical systems have by now been dried out, and the relays and the wiring that failed have been changed. An analysis of the safety in the plant due to flooding and also about the condition of the electrical equipment is ordered [8].

3.3 Hypothetical Scenario

3.3.1 Nuclear Safety Aspects

The hypothetical scenario that will be analyzed in this report is a case in which a break would have occurred in the ECCS-piping at the moment when it was put under the reactor pressure (34 bar), which was higher than its design pressure of 20 bar. Under such circumstances, the single system for adding water to the primary side during a

LOCA and afterwards would not be available. The small break in the primary system may have the potential of uncovering the core, leading to a severe accident in which the following safety related aspects will have to be considered:

- The reactor core is not cooled successfully, core degradation would occur, and D_2-gas which is produced due to the Zirconium oxidation, and fission products could be released to the break.
- The possible retention of fission products (the aerosols, not the noble gases) in the tortuous pathway through the ECCS-piping.
- The break location determines the possibility of fission product retention in the containment, which might be threatened by D_2 combustion.

3.3.2 Break

Given the hypothetical occurrence of a break, the worst assumption is made that the break would have occurred in the part of the ECCS-piping where the diameter was the largest, and where the piping was designed to withstand 20 bar. This assumption originates from the fact that, generally speaking, pipes with a larger diameter are more prone to cracking when pressurized, than pipes with small diameters, see the assumed break location in Figure 4.3.

In consequence of this assumption, the leakage will consist of heavy water, or a two-phase mixture, escaping via all fuel rods in the core (at the time of the incident, the number of fuel elements was 97).

If a break would have occurred elsewhere in the system, water would not escape from the core from all fuel rods in the core (the total break area would decrease) and the mass flow rate of the break would decrease.

3.3.3 Break Location

One possibility was that the break would have taken place inside the containment. For such an event, one should consider also the generation and distribution of D_2-gas, due to steam induced oxidation of Zircaloy. This could lead to D_2 gas combustion in the containment. If the containment remained intact, it would mitigate the environmental consequences of the fission product release from the reactor core.

Since parts of the ECCS-system were located outside the containment, see Figure 2.6, the hypothetical break could have occurred in parts of the piping situated outside the containment. In that situation, primary system-gases and fission products would have been released to the environment, without the mitigating effect of the containment.

Chapter 4

SCOPING ANALYSIS

4.1 Introduction

In the hypothetical scenario to be analyzed (see section 3.3), the assumption is made that a break occurs in the ECCS piping. In consequence, heavy water will leak out of the primary system, which will result in a decrease of the volume and level of the cooling water in the reactor vessel. The analyses in this chapter will try to find an answer to the question; how soon would the leakage of water result in uncovering of the core? And if so, what would the behaviour of the core heat-up look like? An analysis of the rate of fuel temperature rise and the potential metal/water reactions that would take place has also been performed. Possible release of fission products and D_2-gases will be considered, considering a break outside/inside the containment. Different means of cooling the fuel during the blowdown and afterwards, will also be discussed.

4.2 Reactor System Thermodynamics

The first law of thermodynamics forms the basis for estimating the pressure in a leaking vessel;

$$\Delta E = \Delta Q - \Delta W - \Delta E_{leak} = \Delta Q - \Delta H_{leak} \tag{4.1}$$

where:

$$\begin{aligned}
\Delta E &= \text{Change in internal energy [J]} \\
\Delta Q &= \text{Heat transferred to the system [J]} \\
\Delta W &= \text{Work done by the system [J]}
\end{aligned}$$

$$\Delta H_{leak} \quad = \quad \text{Enthalpy leaked from the system [J]}$$

When modeling the changes in a leaking reactor vessel, the terms are expressed by using the core power (P_{core}) and the properties of the leaking coolant.[1]

$$\Delta Q = P_{core}\Delta t \tag{4.2}$$

$$\Delta H_{leak} = (\dot{m}_{leak}x_{leak}h_{stm} + \dot{m}_{leak}(1 - x_{leak})h_{wtr})\Delta t \tag{4.3}$$

$$\Delta E = \Delta(\alpha V \rho_{stm}u_{stm} + (1 - \alpha)V\rho_{wtr}u_{wtr}) \tag{4.4}$$

Where α denotes the volumetric fraction of steam (void fraction), x denotes the mass fraction of steam (quality), V denotes the total volume, and ρ denotes the density.

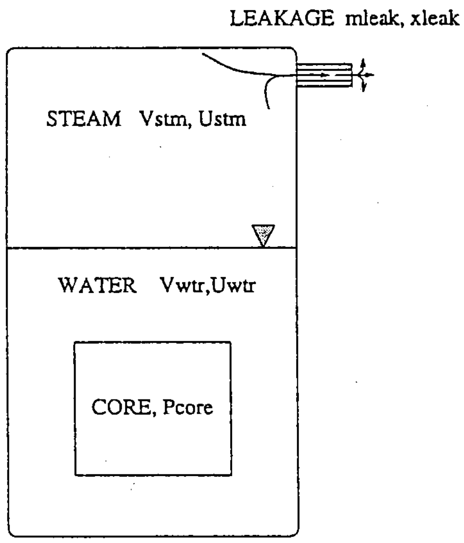


Figure 4.1: Leaking Reactor Vessel

The time discretization of the equations is performed to calculate the pressure inside the vessel as a function of time after leakage has begun. As noted, saturation is assumed all the time during the leakage. In this case, pressure alone determines the properties of the water, which are obtained with functions of Garland and Hand [14] for the water steam equation of state.

After each time step, a certain amount of mass and enthalpy has leaked out from the system, heat has been transferred to the coolant, and the internal energy has changed to E'. The void fraction can be calculated from the total internal energy of the system.

[1]Here the system refers to the mixture of water and steam, which is assumed to remain in thermal equilibrium (saturation) through the event.

$$\alpha' = \frac{E'/V - u_{wtr}\rho_{wtr}}{u_{stm}\rho_{stm} - u_{wtr}\rho_{wtr}} \tag{4.5}$$

where u is the specific internal energy of water and steam.

The corresponding mass can then be calculated;

$$m' = \alpha V \rho_{stm} + (1 - \alpha) V \rho_{wtr} \tag{4.6}$$

The new mass can be based also on the leakage;

$$m' = m - \dot{m}_{leak}\Delta t \tag{4.7}$$

The new pressure is iterated until the equations (4.6) and (4.7) will give the same value of the new mass. The heat from the structures and the influence of the non-condensable gases in the pressurizer are omitted in this calculation.

4.3 Break Flow

Single-phase flow of heavy water, or steam flow through a path with a certain total pressure loss can be calculated using the Bernoulli equation;

$$\Delta p = \sum_i \frac{K_i}{2}\rho_i v_i^2 = \sum_i \frac{K_i}{2}\frac{1}{\rho_i}(\frac{\dot{m}_i}{A_i})^2 = \frac{\dot{m}^2}{2}\sum_i \frac{K_i}{\rho_i A_i^2} \tag{4.8}$$

where:

Δp	=	Pressure drop in the piping [Pa]
K_i	=	Loss factor [-], including frictional pressure drop
ρ_i	=	Density of the flowing liquid [kg/m^3]
v_i	=	Velocity of the flowing liquid [m/s]
$\dot{m}_i$	=	Mass rate of the flowing liquid [kg/s]
A_i	=	Cross-sectional area of the piping [m^2]

By assuming constant density of the flowing liquid (or steam), the serial term becomes independent of water (steam) conditions, and the mass flow rate of the break can be expressed in terms of a reference pressure difference and density.

$$\frac{\dot{m}_{break}}{\dot{m}_{ref}} = \sqrt{\frac{\Delta p_{break}\rho_{break}}{\Delta p_{ref}\rho_{ref}}} \tag{4.9}$$

Such reference values are estimated in the next section. It also has to be noted that the method applied here is approximate, including severe limitations, when considering high velocity flows at high pressure differentials (critical flow, coolant flashing).

In order to relate the primary water inventory to the heavy water level in the reactor vessel, the primary system volumes have to be examined, see Figure 4.2. It is assumed that the most elevated water volumes in the primary system (the pressurizer, the water above the core, and parts of the steam generator volumes) are the first ones to be emptied of water. The total volume of the primary system is 88.7 m^3. The initial void fraction, α, is 0.223. The total heavy water volume above the core is approximately 5 m^3.

This means that a relatively small amount of heavy water has to leak out of the system before the core is uncovered. However, when the system is depressurized, the water in the system begins to boil, and the water level swells.

4.4 Outflow Geometry

As mentioned, the assumption is made that the break would have occurred where the diameter of the piping of the ECCS is largest, and where the piping was designed to withstand only 20 bars. The flowpath via each individual fuel rod to the place where the break would have occurred can be divided into 13 parts (see Figure 2.3 and 4.3). The pressure loss can be calculated for the reference flow (using Equation 4.8), which is the relatively low mass flow rate that the fuel failure detection system has during normal operation ($\dot{m}_{ref}$).

Table 4.1 shows the features of the break flow path, with calculated velocities and pressure drops for the reference conditions, and sections numbered according to Figure 4.3. It has to be noted that the calculation is similar to the design analyses, but no information was found about possible measurements during operational testing of the fuel failure detection system, or the ECCS.

The pressure drop during the break would be the difference in pressure between the reactor and the containment, which is 33 bar at the initiation of the break. When the

Table 4.1: Features of the different sections in the break flow path.

i	$\dot{m}_{ref}$ (kg/s)	d_i (mm)	l (mm)	$(v_{wtr})_i$ (m/s)	K_i (-)	$f\frac{l}{d}$ (-)	Δp (bar)
1	0.85×10^{-3}	1.5	7.0	0.5178	3.5	0.2	0.005
2	3.42×10^{-3}	3.0	40.0	0.5208	1.5	0.5	0.003
3	3.42×10^{-3}	4.0	25.0	0.2930	0.5	0.2	–
4	6.84×10^{-3}	4.0	1.0	0.5859	1.5	–	0.003
5	16.3×10^{-3}	4.5	40.0	0.6498	4.0	0.3	0.008
6	65×10^{-3}	13	70.0	0.5271	3.0	0.1	0.004
7	32.5×10^{-3}	7.5	50.0	0.7919	3.0	0.2	0.009
8	65×10^{-3}	16/12*	300	0.7862	0.5	2.3	0.008
9	65×10^{-3}	18/12*	1400	0.4962	1.5	6.8	0.010
10	65×10^{-3}	9.0	9800	1.010	3.9	26.2	0.170
11	65×10^{-3}	9.0	300	1.010	0.1	0.8	0.005
12	65×10^{-3}	9.0	500	1.010	1.1	1.3	0.014
13	2.102	51	9000	1.107	4.5	2.65	0.041
						$\sum_{i=1}^{13}\Delta p_i$:	0.266

* for $i = 8 : d_h = 4$ mm, and $A_h = 0.89 \cdot 10^{-4}$ m^2; for $i = 9 : d_h = 6$ mm, and $A_h = 1.41 \cdot 10^{-4}$ m^2

reactor pressure decreases, the pressure drop of the break decreases, and the mass flow rate decreases as well, according to Equation 4.9. As the pressure goes down, saturation of the water is reached, and therefore the steam content in the tank becomes larger.

One way of assigning a steam quality of the leaking heavy water is assuming a steam quality of zero ($x = 0$; the leakage consists of water), right after the break has occurred. The quality of the leakage then varies with the level of water in the tank, which is related to the calculated void. One assumption could be that the quality of the steam will change when the saturation pressure of the primary side is reached, but this assumption has not been included into this model.

Another way is to alter the steam quality of the break flow in one step from water ($x = 0$) to steam ($x = 1$). To approximate steam-dominated critical flow through the long flow path and the phase change due to the pressure decrease in the piping of the ECCS, the break flow could be assumed to consist of steam during the whole time of the transient.

The model described here was programmed in FORTRAN, and named TEENAGE (Thermal Equilibrium ENergy calculation for AGEsta).

4.5 Boil-off Level

The water in the core and moderator starts to boil off as soon as the pressure in the primary side reaches saturation pressure (24 bar). After equilibrium between the containment and the reactor vessel has been reached, a pressure difference between the reactor vessel and the containment has to be established in order to press steam out through the long, and thin pipes of the ECCS. The pressure inside the reactor vessel is increasing, as long as no steam is leaking out of the system. When the pressure difference between the containment and the reactor is large enough, steam escapes to the containment, then the reactor vessel is depressurized. This way, the pressure is varying. However, in this model, constant pressure is assumed. The relative water level is 1.0 at the top of the core, and 0.0 at the bottom.

The boil off mass flow is described as follows:

$$\dot{m}_{bo} = \frac{Level \cdot P_d}{\Delta h_{wtr}} \tag{4.10}$$

where:

$$\begin{aligned} \dot{m}_{bo} &= \text{Boil off mass flow rate [kg/s]} \\ Level &= \text{Fractional water level in reactor vessel [-]} \\ P_d &= \text{Decay power [W]} \\ \Delta h_{wtr} &= \text{Latent heat of water [kJ/kg]} \end{aligned}$$

The fractional water level can be expressed like this:

$$(Level)_{n+1} = (Level)_n - \frac{\Delta t (\dot{m}_{bo})_n (v_{wtr})_n}{V_{core}} \tag{4.11}$$

where:

$$\begin{aligned} \Delta t &= \text{Time step [s]} \\ v_{wtr} &= \text{Specific volume of water [m}^3\text{/kg]} \\ V_{core} &= \text{Volume of core [m}^3\text{]} \\ n &= \text{Beginning of time step} \\ n+1 &= \text{End of time step} \end{aligned}$$

The boil-off mass flow rate and level is calculated at each time step. In the model of this section, the pressure is assumed to be constant, 3 bar, which is the maximum pressure in the containment after a LOCA.

4.6 Core Heatup

In order to estimate the amount of time needed for the cladding temperature in the top of the core to reach 1200°C (at which point significant zirconium oxidation begins), a model of the heatup of the core is developed.

The starting point for this model is the time when the level in the core has collapsed, and equilibrium between the containment and the reactor vessel is reached. The pressure in the reactor vessel at this point is 3 bar, which is the maximum pressure in the containment after a LOCA.

By assuming that the part of the core corresponding to the relative water level is covered by water, the boil-off mass flow rate can be expressed, and the mass flow rate per fuel pin is achieved by dividing the boil off mass flow rate with the total number of fuel rods in the core.

$$\dot{m}_{bo} = \frac{Level \cdot P_d}{\Delta h_{wtr}} \tag{4.12}$$

where:

$$\begin{aligned} m_{bo} &= \text{Boil off mass flow rate [kg/s]} \\ Level &= \text{Fractional water level in reactor vessel [-]} \\ P_d &= \text{Decay power [W]} \\ \Delta h_{wtr} &= \text{Latent heat of water [J/kg]} \end{aligned}$$

For the part of the core above water, the velocity and Reynolds number are calculated, and the flow is found to be laminar. For a constant heat flux, and laminar flow, the Nusselt number is:

$$Nu_D = 4.36 \tag{4.13}$$

The heat transfer coefficient is:

$$h_c = Nu_D \frac{k}{D_h} \tag{4.14}$$

where:

$$\begin{aligned} h_c &= \text{Heat transfer coefficient [W/m}^2\text{K]} \\ Nu_D &= \text{Nusselt number} \\ k &= \text{Thermal conductivity [W/mK]} \\ D_h &= \text{Hydraulic diameter [m]} \end{aligned}$$

The temperature difference between the coolant and the surface of the cladding is expressed as the following:

$$\Delta T_{cl-stm} = T_{clad} - T_{\infty} = \frac{q"}{A \cdot h_c} \tag{4.15}$$

where:

$$\begin{aligned} q" &= \text{Power [W]} \\ A = (1 - Level)A_{clad} &= \text{Heat transfer area [m}^2\text{]} \\ h_c &= \text{Heat transfer coefficient [W/m}^2\text{K]} \end{aligned}$$

By dividing the part of the core which is not covered by water into nodes, the surface (or cladding) temperature at different height of the core can be calculated as follows:

$$\begin{cases} \Delta(T_{stm-cl})_i = \frac{q"_i}{h_c A_i} \\ (T_{stm})_{i+1} - (T_{stm})_i = \frac{q"_i}{\dot{m}_{bo} C_p(i)} \end{cases} \tag{4.16}$$

where:

$$\begin{aligned} q"_i &= \text{Power generation in segment i [W]} \\ A_i &= \text{Heat transfer area of core in segment i [m}^2\text{]} \\ h_c &= \text{Heat transfer coefficient [W/m}^2\text{K]} \end{aligned}$$

$$(T_{stm})_{i+1} - (T_{stm})_i = \text{Steam temperature increase in segment i [K]}$$
$$\dot{m}_{bo} = \text{Mass flow rate of boiloff [kg/s]}$$
$$(C_p)_i = \text{Heat capacity of steam in segment i [kJ/kgK]}$$

In this model, the entrance effect of the heat transfer is neglected, and all heat generated in the core is assumed to heat up the steam, ignoring any heat store in the fuel.

4.7 Core Degradation

As a consequence of the uncovering of the core, and the insufficient cooling of the fuel, the temperature of the fuel rises due to the decay heat generation. Two different approaches has been used to estimate the hydrogen production due to zircalloy oxidation during the heat-up transient of the core. In both models, a lumped model of the core is used as described in T. Okkonen,[12].

4.7.1 Heat Balance

An energy balance for the heat up of the core is performed, in order to estimate the order of magnitude of the zircalloy oxidation. The zircalloy oxidation is a significant heat source at high temperatures, and plays an important role in the heat up of the core.

The heat production is equal to heat consumption, which means that the core heatup added to the melting energy is equal to the sum of the decay and oxidation heat;

$$\sum m_i[c_i(T_{m,i} - T_o) + h_{f,i}] = \int P_d dt + X m_{Zr,o} Q_{Zr} \tag{4.17}$$

where:

$$m_i = \text{Mass of material i [kg]}$$
$$m_{Zr,o} = \text{Initial mass of unoxidized zircalloy [kg]}$$
$$c_i = \text{Heat capacity of material i [J/kgK]}$$
$$T_{m,i} = \text{Melting temperature of material i [K]}$$
$$T_o = \text{Initial core temperature [K]}$$
$$h_{f,i} = \text{Latent heat of fusion of material i [J/kgK]}$$

$$
\begin{aligned}
P_d &= \text{Decay power [W]} \\
X &= \text{Oxidation fraction of zircalloy [-]} \\
Q_{Zr} &= \text{Oxidation heat of zircalloy } (6.43 \cdot 10^6 \text{ [J/kg-Zr]})
\end{aligned}
$$

Only the final oxidation fraction is obtained with this method. Information about the timescale of the degradation is also needed, therefore, a transient analysis of the core degradation is performed.

4.7.2 Transient Phase

Zirconium reacts with the oxygen in the steam according to Equation 4.18:

$$
Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + Q_{Zr} \tag{4.18}
$$

Where Q_{Zr} is the oxidation heat of zircalloy (Q_{Zr}(1500K)=6.43 MJ/kg-Zr=4.76 MJ/kg-ZrO_2=16.3 MJ/kg-H_2O=145 MJ/kg-H_2). The temperature dependence of the oxidation heat is not very strong, and is therefore neglected.

The zircalloy oxidation (ZrO_2 formation per area (W [kg/m^2])) is based on the constants used in SCDAP/RELAP5/MOD2, as referred to in [13]. This modeling basically follows the modeling described in [12].

$W = m_{ZrO_2}/A$, ZrO_2 formation per area:

$$
\frac{d(W^2)}{dt} = \begin{cases} 245.2e^{-20060/T} & \text{kg}^2/\text{m}^4\text{s, when } T < 1853 \text{ K} \\ 79.2e^{-16820/T} & \text{kg}^2/\text{m}^4\text{s, when } T \geq 1853 \text{ K} \end{cases}
$$

$W_{H_2} = m_{H_2}/A$, H_2 production per area :

$$
\frac{d(W_{H_2})^2}{dt} = K(T) = \begin{cases} 0.2625e^{-20060/T} & \text{kg}^2/\text{m}^4\text{s, when } T < 1853 \text{ K} \\ 0.08475e^{-16820/T} & \text{kg}^2/\text{m}^4\text{s, when } T \geq 1853 \text{ K} \end{cases}
$$

The heat balance for a lumped core mass can be expressed:

$$C\frac{dT}{dt} = P_d + P_{Zr} \tag{4.19}$$

where:

$$\begin{aligned} C &= \text{Core heat capacity [J/K]} \\ T &= \text{Core temperature [K]} \\ t &= \text{Time [s]} \\ P_d &= \text{Decay power [W]} \\ P_{Zr} &= \text{Zircaloy oxidation power [W]} \end{aligned}$$

Discretization leads to:

$$\begin{cases} (P_d + P_{Zr})\Delta t = C_n \Delta T \\ (W_{n+1})^2 = (W_n)^2 + K(T_n)\Delta t \\ P_{Zr}\Delta t = (W_{n+1} - W_n) A Q_{Zr} \end{cases} \tag{4.20}$$

where:

$$\begin{aligned} \Delta t = t_{n+1} - t_n &= \text{Time step [s]} \\ \Delta T = T_{n+1} - T_n &= \text{Temperature step [K]} \\ W &= \text{Hydrogen production per area [kg/m}^2\text{]} \\ A &= \text{Area [m}^2\text{]} \\ Q_{Zr} &= \text{Heat of Zr oxidation reaction [J/kg-Zr]} \\ n &= \text{Beginning of time step} \\ n+1 &= \text{End of time step} \end{aligned}$$

Equation 4.20 is combined into a second-order equation for the cumulative hydrogen production after each time step (W_{n+1});

$$(W_{n+1})^2 + \frac{K(T_n)}{P_d} A Q_{Zr} W_{n+1} - \{(W_n)^2 + \frac{K(T_n)}{P_d}(A Q_{Zr} W_n + C_n \Delta T)\} = 0 \tag{4.21}$$

$$\Rightarrow \; W_{n+1} = W_{n+1}(W_n,\; T_n,\; \Delta T,\; P_d) \tag{4.22}$$

A temperature step is selected, and Equation 4.22 gives the cumulative hydrogen production ($W_{n+1} - W_n$), and Equation 4.20 the corresponding oxidation power (P_{Zr}), and time step (Δt).

The heat capacity of the core mass (C_n) is calculated at the beginning of each time step.

Zircaloy oxidation demands a certain steam availability to become fully developed. In case of a LOCA, the water level in the core decreases, and the supply of steam is limited. This limits the oxidation power below a certain level. In such a case, the zircalloy power, and the time step, are calculated as follows;

$$\begin{cases} P_{Zr} = C_{STARV} \cdot \frac{P_d}{\Delta h_{wtr}} \\ \Delta t = \frac{C_n \Delta T}{P_d + P_{Zr}} \\ W_{n+1} = W_n + \frac{P_{Zr} \Delta t}{A Q_{Zr}} \end{cases} \tag{4.23}$$

Where C_{STARV} is a constant, which describes how much of the boil off mass flow rate is assumed to be used for zirconium oxidation.

The total amount of hydrogen production is calculated. The hydrogen fraction is gained by comparing the amount of hydrogen to the total volume, or parts of the containment. This way, the risk for hydrogen combustion can be estimated. The flammability limits for hydrogen in air is between 5 % – 15 %, depending on how wet the air is.

4.8 Alternative Systems That Could Possibly Have Been Used for Cooling the Core

The ECCS that could have been failed was designed to be used in the event of a LOCA. If the ECCS had failed, the question is what other system could have been used to add water to the primary system?

At the time of the incident, the supply of heavy water in the plant was limited to approximately 100 m^3, including the primary side water. Also, if the water of the blowdown ends up into the containment, any condensation would end up in the drainage system which at this point was overfilled with water from the flooding (see Section 3.2). Also, the drainage basin was connected to the ECCS. Therefore, the recirculation system that was to be used during an accident, would not have functioned.

Two systems existed for filling and emptying of the primary side water:

- The System for Filling and Emptying the Primary Side Water at Low Pressure, P90. This system was at the time of the incident, and during normal operation, connected to the heavy water storage tanks, with a total storage capacity of 75m^3. The system was constructed to add water to the primary side with a mass flow rate of 1 kg/s or 7 kg/s. The higher mass flow rate could be used only if no need for cleaning the water before injecting it existed. The water was injected to the annulus flow (see Section 2.4: The Primary Side). This system has an operational pressure of 7 bar and a design pressure of 20 bar [1].
- The System For Adding Heavy Water To the Primary Side During Operation, P88. This system was primarily constructed to recover the heavy water that escaped from the primary side (leakage in valves, etc.) during normal operation. The stored amount of heavy water in this system was only 2.8 m^3 and the maximum mass flow rate that the system could add to the primary side is 0.5 kg/s. The water was injected to the main pump. The system has an operational pressure of 40 bar and a design pressure of 59 bar [1].

By the end of 1969, a total assessment of the heavy water inventory in Ågesta was made [10], see Table 4.2. It can be noted that a relatively small amount of heavy water was stored in the storage tanks, and in the laboratory. The quantities of heavy water is converted to 100% D_2O, the weight is calculated for the pressure 1 bar.

Table 4.2: Heavy water inventory in Ågesta.

	D_2O (kg)
Primary system	73,264
Refuelling machine	500
Storage and laboratory	2,141

This examination shows that if the ECCS piping had ruptured in the incident, the only system for adding water to the primary side would have been lost. Therefore, the main interest of an ECCS break scenario are related to the accident time progression (fuel failures) and the mitigation of radioactivity released (break location).

One can assume that the system for filling and emptying the primary side of its heavy water during low pressure could be used for adding water after the point in time when the pressure in the primary system had dropped below 7 bar. In order to operate the system in this situation, it would have to be connected to an additional source of light water, a task that had not been contemplated prior to the incident.

4.9 Analysis Results

4.9.1 TEENAGE

To get a general picture of the behaviour of the different parameters, the steam quality of the blowdown was assumed to vary in three ways;

- (1) The leakage consisted of steam all the time during the blowdown (simulating steam-controlled critical flow at the end of the break),
- (2) The steam quality is varied linearly from zero to one from the top of the vessel to the time when only two thirds of the vessel volume is filled with water.
- (3) The quality of the leakage is kept at zero (the leakage consists of water) until a certain level in the vessel is reached; at this point, the quality is set to one instantaneously (the leakage consists of steam).

The time elapsed up to the point when the reactor vessel is half empty is plotted in Figure 4.4 ($y = 1.0$ denotes the top, and $y = 0.0$ denotes the bottom of the core). This time is strongly dependent on the quality assumptions of the outflow; 11000 s for the case when the leakage consists of only steam, 3620 s for the case when the steam quality of the leakage is gradually increasing, and 760 s for the case when the leakage is instantaneously turned from water to steam when a certain point in the reactor vessel is reached.

Figure 4.5 shows the pressures reached at the point when the heavy water is covering half of the core (level = 0.5) for the three different cases. The pressures for the different cases are 1.43 bar, 4.5 bar, and 18.5 bar, respectively. The times for the pressure to reach 2 bar for the different assumptions of the outflow steam quality are; 8120 s, 7411 s, and 6070 s respectively for the different cases.

In Figure 4.6, the mass flows rates for the different cases are shown. This property varies according to the assumptions made regarding steam quality of the break mass flow.

4.9.2 Boil-off Level

The boil-off level as a function of time is presented in Figure 4.7. This level has been calculated for different levels of decay heat, representing the different times required to uncover the core down to a relative level of 0.5, as explained in the section above.

The boil off is a very slow process because of the relatively large moderator/fuel volume ratio, 16.4, and the relatively low surface heat flux of this reactor. The boil-off mass flow rate varies from 0.35 down to 0.08 kg/s. The total time for the mass to boil-off from a relative level of 50% down to 10% of the core is approximately 200,000 – 300,000 s (55 – 83 hours), depending on the decay heat.

4.9.3 Core Heatup

The model describing the heatup of the core made use of the total flow area, and the hydraulic diameter of a fuel rod. The temperatures in the top of the core were calculated for two different decay heat levels, 1%, and 0.67%.

The result of the calculation is presented in Figure 4.8, showing that the relative water level in the core has to drop via boil off to approximately 10% of the total height of the core before the cladding temperature in the top of the core starts to rise significantly, and down to 5% until the temperature of 1200°C is reached. The levels are hardly dependent of the decay heat assumed. It also shows that the steam temperature is independent on the decay heat.

The result shows that a large amount of water will have to boil off until significant zirconium oxidation starts.

4.9.4 Core degradation

Heat Balance

In order to use the heat balance, the parameters of Ågesta HPWR are provided in Table 4.4. The effects of oxidation of steel is neglected, because of the relatively small oxidation heat of steel.

The decay heat (P_d), and the fuel melting temperature (T_{fr}) were varied, and the zircalloy oxidation fraction was calculated, using Equation 4.17. The result is shown in Table 4.5. The results tell us that the oxidation fraction of zircalloy increases with smaller decay heat and larger melting temperature. This means that the more restricted the zirconium oxidation is, due to steam starvation and low decay heat, the larger the zircalloy oxidation fraction will become. Because of the fact that the zirconium oxide has a higher melting point than the zirconium, the melting is considered to take a longer time, and the oxidation fraction increases. In Table 4.5, the zircalloy oxidation fractions for different assumptions of the decay heat is presented. The larger the melting temperature, and the lower the decay heat, a higher zircalloy oxidation fraction is obtained from the heat balance.

Transient Phase

The core degradation is limited by the supply of steam in the core, which is dependent on the water level. As could be seen in the analysis of the core heatup, the water level had to decrease down to 5% – 10% until the temperatures in the top of the core reaches 1200°C, at which point significant zirconium oxidation starts. Therefore the constant that describes the steam starvation (C_{STARV}) is not assumed to be higher than 0.05.

The availability of steam is largest near the water line, and most of the oxidation takes place there. In this lumped model, the local differences in the oxidation fraction is not modeled. Also, the lumped mass in this model will melt at a certain melting temperature. Of course, some parts of the core will melt earlier than others, therefore decrease the total area of the zirconium. The melted parts of the core also create an increase in the steam flow, as the melt comes in contact with water. This effect is neglected in this model.

The results of this analysis are presented in Figure 4.9 and 4.10. In the figures, the behaviour of the zirconium oxidation for different levels of the steam starvation are shown. In all curves, the decay heat is assumed to be 1% of the nominal power, but the starvation is changed. The most possible behaviour of the starvation is the case when $C_{STARV} = 0.05$, which represents the level of water in the core when the cladding temperature reaches the point where significant zirconium oxidation begins.

4.10 Discussion

The assumptions made regarding the quality of the mass flow rate do not at all describe the most probable course of the water level change during the transient. Instead, the water level would most likely be following case 3 in the first phase of the transient, while the water level still is above the location of the inlet to the break flow geometry. Later, it would follow case 2, where part of the mass flow rate consists of steam. Finally, the mass flow rate would consist entirely of steam, and would, therefore, follow case 1, as shown in Figure 4.4.

Furthermore, the subcooling of the coolant in the primary side is not modeled. When the pressure reaches saturation, flashing occurs, and steam is formed. The saturation pressure at the prevailing subcooling, at the top of the core, in Ågesta was 24 bar. Also, in the break flow geometry, with very thin pipes, no critical flow restrictions have been taken into account in the modeling described in this chapter. The critical flow restricts the mass flow rate out of the break.

In the analysis of the heat-up of the fuel, it is noted that the water level has to reach down to a relative level of around 10% – 5% before significant heat-up of the fuel begins. According to the analysis of the boil-off, this takes 200,000 seconds (55 hours) or more, which is a long time. During this time several courses of action could be taken to prevent the water level in the core from decreasing. For example, the system for filling and emptying the primary side water during low pressure could be connected to an external light water source. Therefore, it is most probable that the temperature of the fuel never will reach the point where significant zirconium oxidation begins. This modeling of the heat up of the core has been included to consider all the possible scenarios.

Figures 4.4 and 4.5 show that the collapsed level, and the pressure in the reactor vessel depend of the steam quality of the break mass flow. This parameter is not calculated, but assumed. In order to calculate this parameter, and the initial dynamical phase of the transient, RELAP5/MOD3.1 could be used. This would give a more accurate picture of the behaviour of the initial phase of the transient, and would also give a more reliable prediction of the level and pressure in the reactor vessel.

Also, possible steam formation in the primary side steam generator piping (the location in the primary side piping with the lowest temperature and pressure) during the blowdown, and other dynamical limitations, such as critical flow, should have a large impact on the mass flow rate out of the break. Other probable events considered to have impact on the mass flow rate of the break were; natural circulation, heat transfer from the secondary side to the primary side, and level swelling in the reactor vessel. Therefore, it was decided that a RELAP5/MOD3.1 model of the Ågesta PHWR should be developed.

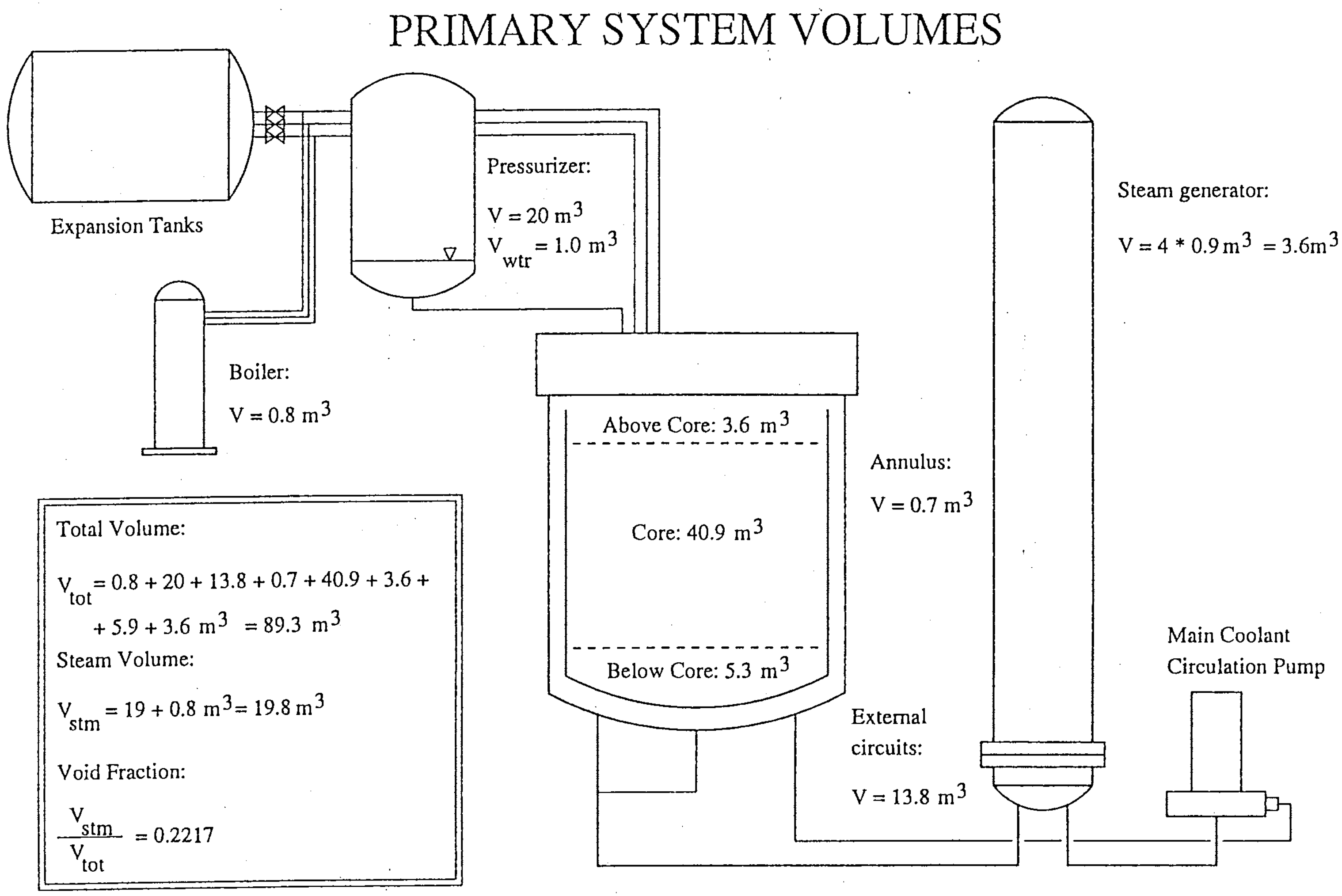


Figure 4.2: The Water Volumes of the Primary System.

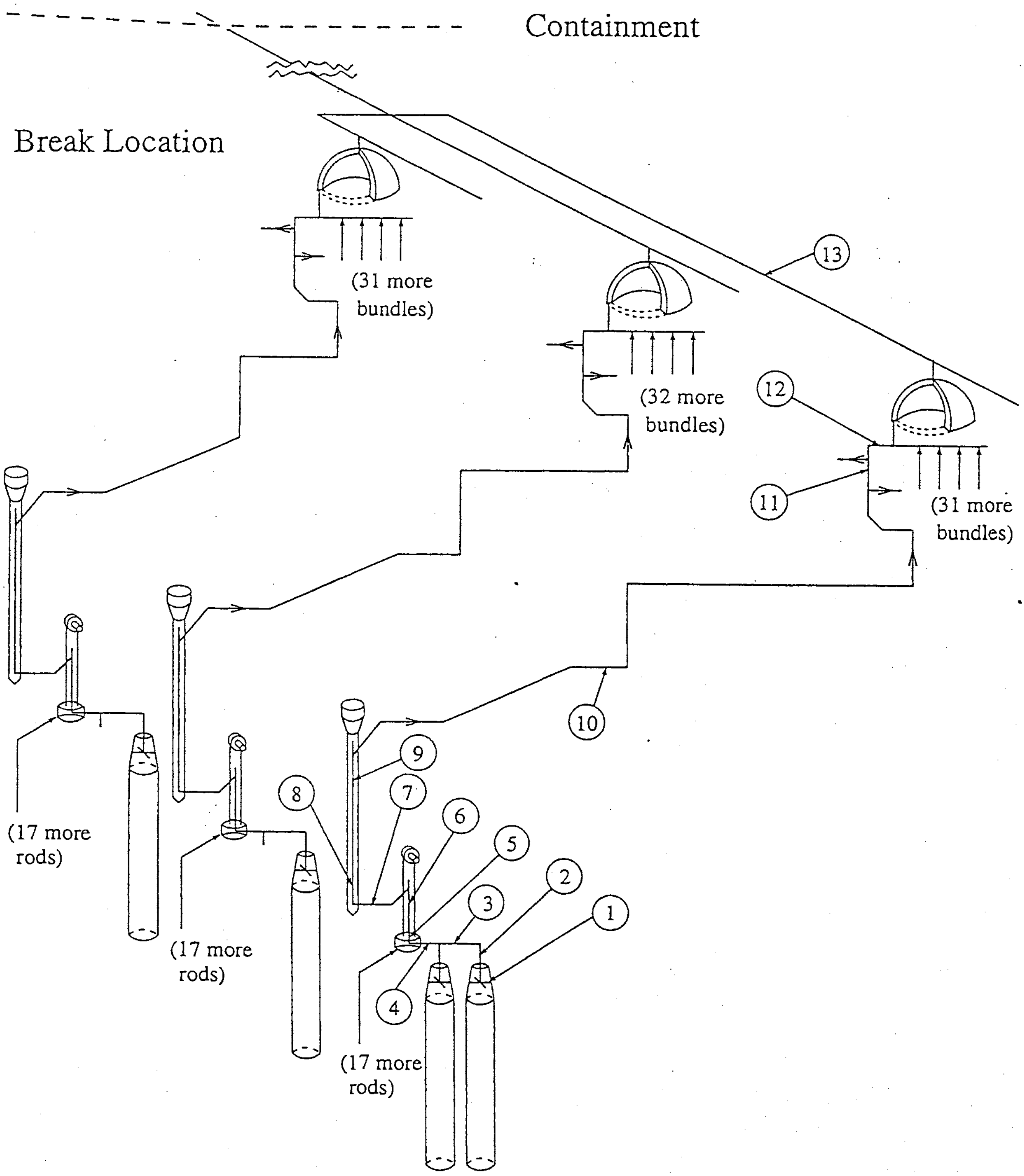


Figure 4.3: Break Flow Geometry (Corresponding To Table 4.1)

Table 4.3: TEENAGE Results.

CASE	1	2	3
Time to Uncover half of core (s)	11,000	3,620	760
Pressure when level = 0.5 (bar)	1.43	4.5	18.5
Time to reach pressure = 2 bar (s)	8,120	7,410	6,070

Table 4.4: The materials in the core of Ågesta.

Material	Mass (kg)	c_p (J/kgK)	h_f (kJ/kg)	T_m (K)
UO_2	12610	400	274	T_{fr}
Zr-clad	$(1-X) \cdot 2300$	370	225	T_{fr}
ZrO_2-clad	$X \cdot 3105$	540	707	T_{fr}
Zr-can	$(1-X) \cdot 900$	370	225	2100
ZrO_2-can	$X \cdot 1215$	540	707	2100
C (MJ/K)	$6.255 + 1.053 \cdot X$			
H_f (GJ)	$4.175 + 2.334 \cdot X$			

Table 4.5: The Oxidation Fraction for the Zircaloy in the Core.

P_d (%)	$\int P_d\, dt$ (GJ)	T_{fr} (K) 2100	2600	3100
0	0	52.5	76.4	94.8
1	0.975	47.3	71.2	89.4
2	1.95	42.1	65.9	84.1

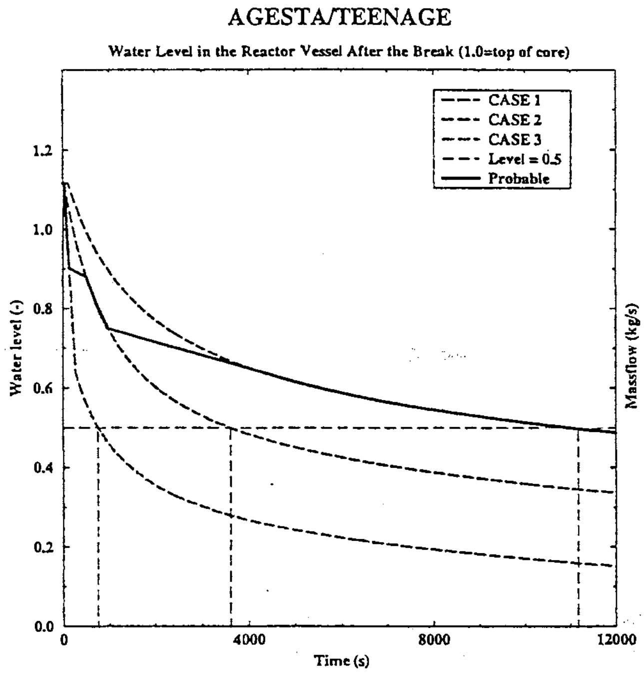


Figure 4.4: TEENAGE: Levels During the Blowdown.

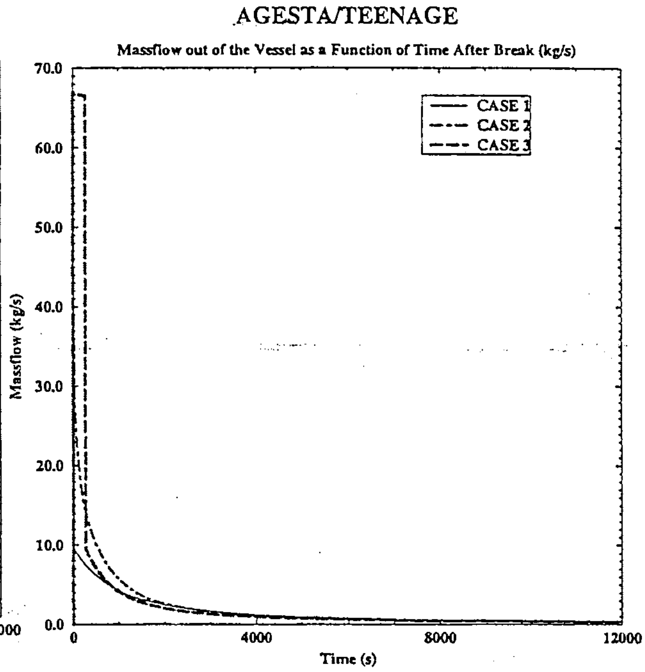


Figure 4.6: TEENAGE: Massflow of Blowdown.

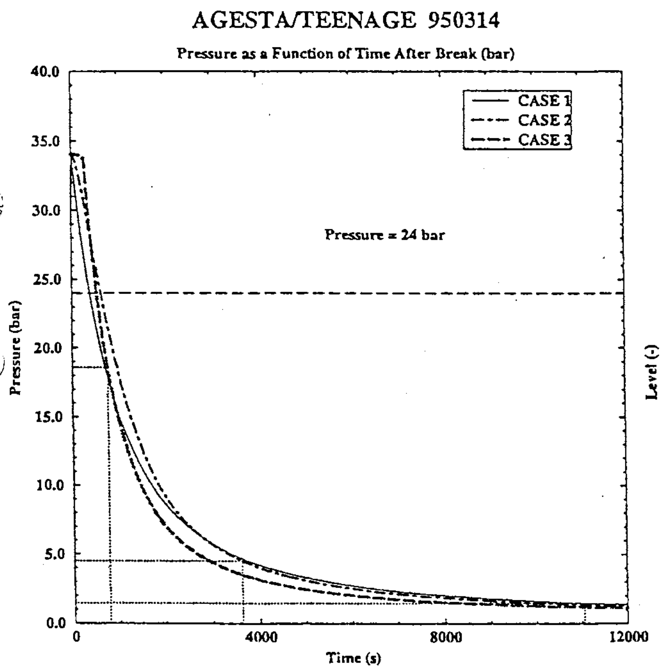


Figure 4.5: TEENAGE: Pressures when Level = 0.5

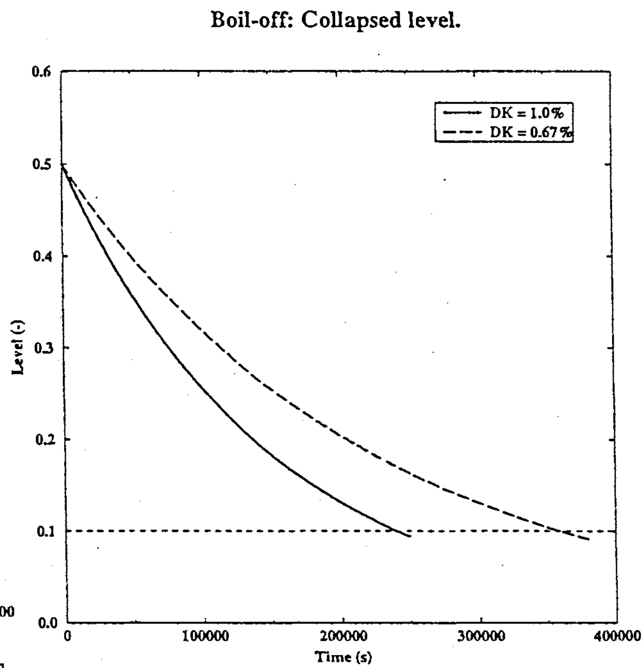


Figure 4.7: Boil-off: Collapsed level.

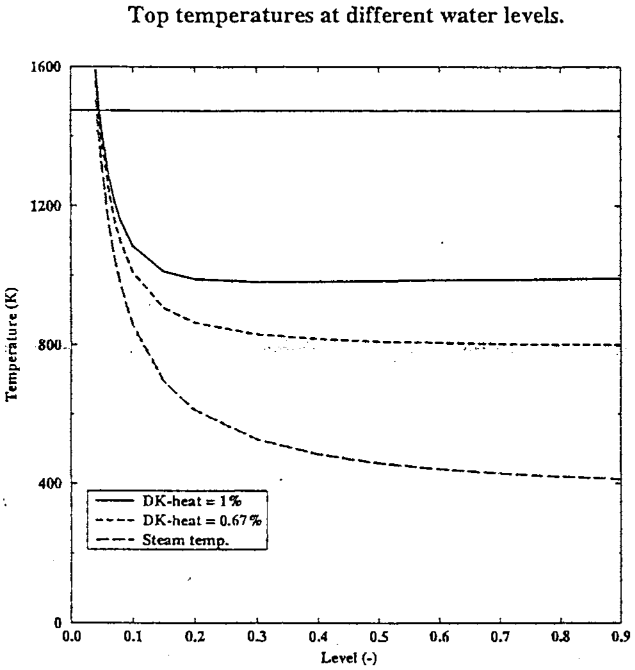


Figure 4.8: Heatup: Temperatures in top of core.

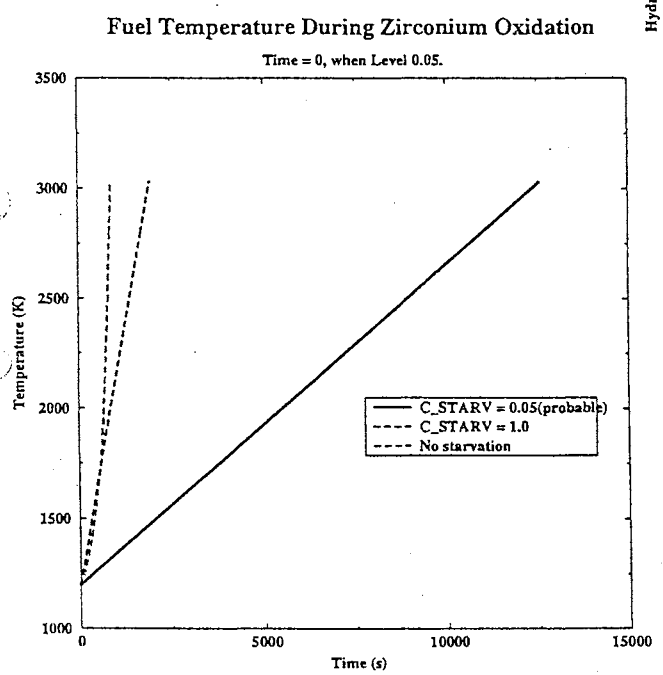


Figure 4.9: Temperature rise due to zirconium oxidation.

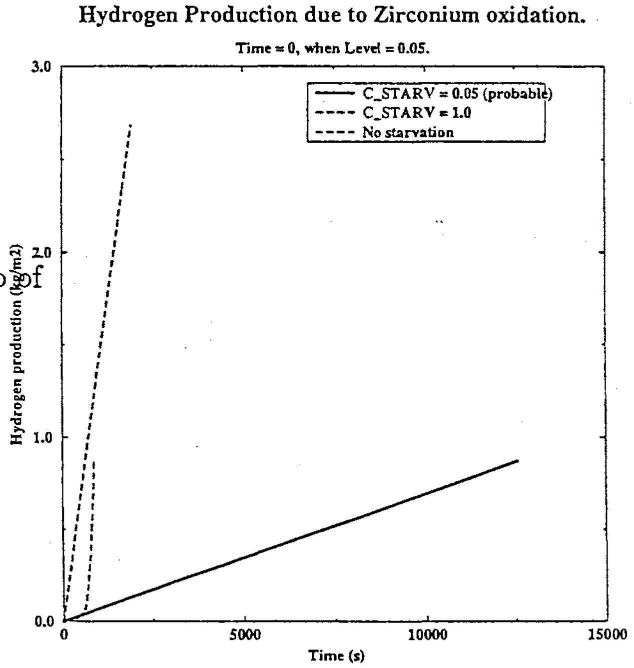


Figure 4.10: H_2 - production due to zirconium oxidation.

Chapter 5

DYNAMICAL TWO-PHASE FLOW ANALYSIS

In order to simulate the dynamics in the primary side during the blowdown, an input deck of the Ågesta reactor was developed for the RELAP5/MOD3.1 code.

5.1 Code Description

The RELAP5/MOD3.1 code has been developed for best-estimate transient simulation of light water reactor coolant systems during severe accidents [15]. It has the capability of running both small and large LOCAs, as well as operational transients. It models the coupled behaviour of the reactor coolant system and the core. It has a general two phase flow model well validated through several experiments, and is today the most widely used code of this kind.

5.2 Input Description

In order to use RELAP5/MOD3.1, a fairly advanced nodalization of the plant must be developed. The information about the Ågesta design was taken mainly from References [2] and [1]. The design drawings of the plant were used to measure the lengths of the piping, and the diameters were found in [1]. Some details of the plant were discussed with former staff members of the plant, and a visit to the plant was made. Both the primary and the secondary side of the plant were modeled.

Some of the information needed for the input was impossible to obtain, despite several visits in archives, and aid from helpful people at SKi and Vattenfall in Stockholm. Therefore, material from inputs of other modern plants were used. Because of the unique design, and the small size of Ågesta, alterations were made in order to develop physically reasonable information. When the code required detailed information that was not in our possession, simplifications were made. When derived and/or assumed information of this kind is used, it is carefully noted in the input. The potential influence on the results was also considered.

The nodalization was made according to the guidelines in [15]. A complete listing of the RELAP5/MOD3 input of Ågesta can be examined in Appendix A.

5.2.1 Hydrodynamic Model

The reactor and the four main primary loops, including the hot leg, the primary side of the steam generators, the main pump, and the cold leg were all lumped into a single loop of components, see Figure 5.1. All loss coefficient was determined using Idelchick: Handbook of Hydraulic Resistance, [16].

The hot leg (CCC(component number)=100) has a diameter of 275 mm, and a total length of 12.4 m. The inlet plenum of the steam generator (CCC=103) is connected to the steam generator primary side piping (CCC=105) by a spreader (junction 104) with a loss coefficient of 43.0. The steam generator primary side consists of 1994 parallel pipes with a diameter of 8.2 mm. The average length of the tubes is 8.0 m.

The steam generator outlet plenum (CCC=107) is connected to the first cold leg section (CCC=110, the pipe connected to the suction side of the main pump), it has a diameter of 275 mm and a length of 7.3 m.

The information about the main pumps in Ågesta was very brief, and insufficient for use with the advanced pump model parameters required in RELAP5/MOD3. Therefore, the simplified in-built model in RELAP5/MOD3 was used for the main pump (CCC=120), and the only thing that was altered was the rated head, and the moment of inertia of the pump. When tripped, the pump coasted down in less than 100 s.

The outlet of the main pump is connected to the cold leg (CCC=121), which is 10.9 m long, and has two different diameters, 275 and 225 mm. The annulus flow pipes are connected to the cold leg, but due to the relatively small mass flow in the annulus flow, this was not modeled.

The cold leg is connected to the inlet plenum of the pressurized vessel (CCC=312).

This is divided by the spreader. The two parts have approximately the same volume, 1 m^3, but different areas and lengths. The spreader has a loss coefficient of 14.89.

The fuel channel (CCC=317) consists of 97 parallel flow areas. The total area is 0.387 m^2, and the length is 3.14 m. The hydraulic diameter is 0.110 m. The top of the fuel elements, is connected to the ECCS, and the flow comes out of the system at the assumed break in the ECCS during the transient. Two junctions (junctions number 315 and 319), at the bottom and the top connect the fuel elements to the moderator (CCC=330).

The upper plenum (CCC=320) is connected to the top of the fuel element. This volume is also connected to the pressurizer (CCC=400) and the moderator (CCC=330). It has an diameter of 4.21 m, and a height of 0.45 m.

The moderator (CCC=330) has the same length as the fuel elements (CCC=317), an area of 13.0 m^2, and a hydraulic diameter of 1.1166 m. It is connected to the outlet of the vessel (CCC=332), which has an volume of 0.49 m^3, and a height of 0.7 m. It is connected to the hot leg and completes the loop.

The pressurizer (CCC=400) consists of a tank and a surge line. Their total volume is 22.5 m^3. The total water volume in the pressurizer tank and surge line during normal operation is 3.5 m^3. The Safety Relief Valve (SRV) is connected to the top of the pressurizer.

The break flow geometry is described in Section 4.4 'Break Flow Geometry', and has been modeled according to Figure 4.3. Some simplifications of this complex system had to be developed in order to keep the computation time reasonable (see Section 5.4). The break flow geometry is connected to the fuel channel and the containment by break valves, which are valves that open simultaneously.

The total volume of the primary side is 86 m^3, with a total weight of D_2O of 63.2 tonnes during normal operation.

The secondary side was also modeled, as described below.

The separator (CCC=700) separates the steam from the water. The water falls back from the separator to the liquid (CCC=710), which is a circular annulus outside the volume that is in direct contact with the boiling region of the steam generator tubes (CCC=715). The liquid falldown has an area of 1.2 m^2, a hydraulic diameter of 0.1 m and is 5 m high. It is connected to the boiling region volume by a collector (CCC=712). The boiling region of the steam generator (CCC=715) has a area of 6.79 m^2, and a height of 4 m. A hydraulic diameter which is equal to the tube to tube spacing (4 mm) in the steam generator is used, according to [15].

At the top of the boiling region, the feed water line is connected. The feed water tank is modeled by a time-dependent volume (CCC=750), set to have a constant pressure and temperature, and a time dependent junction (CCC=751), which is set to adjust the inlet mass flow rate as a function of the void fraction of the volume above the boiling region (CCC=717), and also the level in the secondary side. When the level rises, the void fraction decreases. In order to keep the level constant, the feed water flow is decreased as the void fraction decreases.

Above the séparator is the steam dome (CCC=720) connected to the steam line. This is modeled by a junction (CCC=760) and a time-dependent volume (CCC=761), set to have a constant pressure equal to that of the feed water volume. Connected to the steam dome is also the safety valve, which opens at 30 bar, and closes at 28 bar.

The total volume of the secondary side is 87.8 m^3, and has a weight of 33.1 tons during normal operation.

5.2.2 Heavy Water Properties

In the RELAP5/MOD3 model of Ågesta PHWR, both the primary and the secondary sides are considered to have heavy water as coolant. By comparing the properties of the two fluids, the impact on the result could be assessed. The results of the investigation are shown in Table 5.1.

The properties were compared for three different temperatures, 100°C, 200°C, and 240°C. D_2O has higher densities, higher kinematic viscosity, lower enthalpies, and lower thermal conductivity for all three temperatures. For the conditions in the secondary side during normal operation, the latent heat ($h" - h'$) for D_2O is 10% lower than for H_2O. The figures in Table 5.1 are taken from [17] and [18].

It is assumed that the differences of the properties of H_2O and D_2O are of minor significance compared to the other uncertainties in the input and in the applicability of RELAP5/MOD3 models and correlations.

5.2.3 Heat Structures

A number of heat structures are included in the model. Before the heat structures were modeled, an estimation of the contributions from the different heat structures was performed. Because of the large reactor vessel, and the low power of the core, the contribution from the internal structures could be significant, and was included in the model.

The most important heat structures are the fuel rods (1317) and the primary side steam generator tubes (1105). The fuel rods are divided into 6 axial structures, each having 6 radial nodes (including the cladding, and the gap between the fuel and the cladding). The steam generator tubes are divided into 8 axial structures, each having 2 radial nodes.

The point kinetics model included in the RELAP5/MOD3 requires various values of the reactivity feedback that were not found. It was decided that instead of using the point kinetics model, a time-dependent heat source was assigned to the fuel rods.

A general table (General Table number 999) describing the decay heat curve was implemented in RELAP5. The decay heat curve was taken from ANSI 5.1, and was calculated with a full operating power of 65 MW. The scram time was considered to be 6 s, and therefore the fission power was linearly decreased from 60.7 MW to 0 MW and added to the decay heat, 4.2 MW initially.

There were 97 fuel elements in the core at the time of the incident, each consisting of 19 fuel rods. The total heat transfer area to the cooling water is 319.0 m^2.

The heat structures that connect the primary to the secondary side consist of a total of 7976 stainless steel tubes, with an inner diameter of 8.2 mm, and a thickness of 1 mm. The total heat transfer area on the primary side is 1644.0 m^2, and 2044.6 m^2 on the secondary side.

Also the following internal structures were modeled: the fuel channel between the coolant (CCC=317) and moderator (CCC=330), the pressurized vessel tank, the piping of the hot leg, the piping of the suction side of the cold leg, the pressurizer tank, and the surge line piping.

5.2.4 Trip Logic

The following trips have been included into the RELAP5/MOD3 Ågesta model;

- The scram
- The pump stop signal
- The closing of the feed water inlet and steam outlet valves of the secondary side

- The Safety Relief Valve of the primary side, which is connected to the top of the pressurizer, and has an opening pressure of 42 bar, and a closing pressure of 41 bar.
- The Safety Valve of the secondary side is connected to the top of the steam generators, and has an opening pressure of 30 bar, and a closing pressure of 28 bar.
- The break valves are tripped simultaneously.

These trips can be activated at different times, and separately. They all play a role in the Ågesta incident.

5.3 Steady State Analysis

Before the transients were run, a steady state was searched with the input. It was accomplished in three steps. First, the primary side was run without any heat structures to establish the proper hydraulic condition. The plant was designed to have a larger primary mass flow rate (1200 kg/s) than during the time of the incident (1020 kg/s). In Table 5.2, a comparison between the values from [1], the technical description of Ågesta PHWR, and the values of the steady state run for the larger mass flow rate.

Also, the secondary side was run first separately, adding heat to the steam generator tubes. This way, the function and the steady state properties were established.

Third, the two parts were connected and run together. The secondary pressure was altered in order to adjust the temperatures in the primary side. In Table 5.3, the steady state parameters for the complete system are shown. The rated head of the pump was also changed slightly, from the first steady state, in order to obtain the correct mass flow rate in the primary circuit.

5.4 Transient Analysis

After the steady state analysis was done, the break flow geometry, see Figure 4.3, was implemented to the code. The small junctions and diameters made the Courant limit (the time it takes for the fluid to completely pass a volume) very small. Simplifications were made in order to increase the speed of the calculation. The number of control volumes and junctions was reduced by lumping, so that the lengths and volumes were increased. The node lumping was found to have an insignificant effect on the results.

The break is modeled with two trip valves that open on the same signal, one at the junction between the containment and the break flow geometry (the point of the break), and one at the top of the fuel elements, where the outlet to the break flow geometry in the core would have been situated. When the valves open, the vessel is immediately connected to the containment, and the coolant starts to flow out of the primary system.

Several strings of events were considered for the transients that were run.

In the base case, the break opening was assumed to cause immediate reactor scram, tripping of primary pumps, and secondary feed water, as well as closure of secondary steam lines.

In addition to the base case, a number of alternative transients were run to establish the importance of different features in the model:

- (1) The junction connecting the top of the fuel elements to the top of the moderator was excluded.
- (2) The pump was not tripped at the moment of the break.
- (3) The pump, the feedwater line, the steam outlet was not tripped, and the reactor was not scrammed.
- (4) The area of the break was set to a 5" break, which is equivalent to the total amount of all the small pipes above the fuel rods through which the reactor coolant flows to the break location.
- (5) The break flow outlet location was altered.
- (6) The system for adding water during low pressure was implemented, adding water to the top of the vessel when the pressure reaches 7 bar.

5.5 Base Case

The pressure in the pressurizer, in the secondary side, and in the containment, the mass flow rate of the blowdown, the velocities and void fractions of the break flow geometry, and the pressure loss in the breakflow geometry are shown in Figures 5.2 – 5.6. The mass flow out of the reactor vessel into the containment is restricted by critical flow until $t = 4000\ s$. At that time, a small increase in the mass flow rate can be noticed, as seen in Figure 5.3, and also a sudden decrease in the pressure difference between the containment and the outlet of the break flow geometry, as seen in Figure 5.6.

5.5.1 Coolant Flashing

Coolant flashing occurs due to the de-pressurization of the coolant in the reactor vessel right after the break. When the pressure in the reactor vessel reaches the saturation pressure for the coolant, the coolant saturates.

5.5.2 Natural Circulation

After the pump has coasted down ($t \approx 100$ s after the break), natural circulation in the primary side piping is established due to the temperature differences in the different parts of the piping. The mass flow rate of the natural circulation is $m_{sc} = 80$ kg/s, or around 8% of the mass flow rate during normal operation.

5.5.3 Steam Void Formation in Steam Generator Primary Piping

In the first 800 s of the transient, water and steam are leaking out of the system. During this time (not including the coast-down of the pump) the pressure is decreasing slightly, see Figure 5.2. As the water level in the reactor vessel reaches a point below the outlet of the break, the void fraction in the inlet and outlet of the break geometry increases, and the pressure in the reactor vessel decreases, continuing to a value of ≈ 5 bar in ≈ 6000 s, see Figures 5.5, and 5.2.

After 1000 s, steam is formed in the primary side piping of the steam generator as the pressure and temperature of the coolant reaches the secondary side conditions (20 bar, saturation). Furthermore, the direction of the heat transfer between the primary and secondary side is altered at the same time. The heat transfer rate from the secondary to the primary side is not as efficient, due to the steam in the primary side piping in the steam generator which does not transfer the heat as efficiently as water.

Also, as heat is transferred from the secondary side to the primary side, coolant saturation and steam expansion occur, thereby increasing the pressure and lowering the water level in the hot and the cold leg.

When the water level in the steam generator hot and cold leg is pressed down, the water level in the reactor vessel increases. When the level is low enough, steam slugs can escape, which causes a decrease in pressure, as well as an increase in the water level in the steam generator primary side piping. The level oscillations of the coolant in the reactor vessel causes the instabilities in the break mass flow rate between 1000 and 1600 s. The temperatures in the primary and the secondary side of the steam generator are presented in Figure 5.7.

The total volume of the steam generation in the primary side due to the heat transfer from the secondary side of the steam generator is approximately 12 m^3, which is equal to a level rise in the reactor vessel of approximately 0.5 m.

In the base case, the secondary steam pressure is not lowered by dumping of the secondary side steam. If the secondary side were to be dumped, the steam generator pressure and temperature would decrease. The steam in the primary side piping would be cooled down, and the water level in the reactor vessel would decrease.

5.5.4 Critical Flow

The flow in the break flow geometry is restricted by critical flow in different sections until $t = 4000$ s after the break. After 4000 s, the reactor pressure decreases below the value required to maintain the critical flow in the junction which connects the pipe break with the containment. The pressure difference between the containment and the outlet of the break flow geometry instantaneously disappears, see Figure 5.6.

5.5.5 Core Heatup.

After approximately 2700 s, water level decreases to uncover the fuel bundles and the fuel temperature at locations near the top of the core slowly begins to rise.

In Figure 5.8 the cladding temperature rise, as calculated with RELAP5 is presented. The temperature gradient for the last 200 s of the transient calculated is around 0.1 K/s. The oscillating behaviour depicted is due to the variations in the calculated heat transfer coefficient by the RELAP5/MOD3 code as the water level oscillates in the top nodes of the fuel rods.

5.5.6 Extension of RELAP5

After 4000 s, the break mass flow rate is no longer critical, and consists entirely of steam. Further calculation of the pressure and the mass flow rate can be performed using the thermal equilibrium equation, combined with the Bernoulli equation (TEENAGE) described in Chapter 4. This way, the time to reach equilibrium between the containment and the reactor vessel can be calculated. This extension is showed in Figures 5.16, and 5.17. According to this calculation, equilibrium between the reactor vessel and the containment is reached after approximately 16,000 s.

In Figure 6.1 and 6.2, the calculated collapsed levels during the critical flow part of the transient, and after equilibrium is reached are shown, respectively.

5.6 Comparisons Between Base Case Transient and Some Variations

5.6.1 Accident Management

The only system that could have been used to add water to the primary system during the transient is the system for filling and emptying the primary system during low pressures (see Section 4.8). It has a maximum filling capacity of 7 kg/s. The system was connected to the storage tanks, and its maximum operating pressure was 7 bar. Water is added to the annulus flow, which is connected to the top of the moderator.

In the RELAP5 calculational model, the water is added to the top of the moderator at a rate of 7 kg/s. The added water has a temperature of 60°C. The system is assumed to be actuated as the pressure at the top of the moderator falls below 7 bar.

In Figure 5.9, the temperature of the cladding is shown for the base case and for the case where light water is added after 4200 seconds, i.e. when the primary system pressure reaches 7 bar. This figure indicates that the cladding temperature decreases shortly after the water is added.

5.6.2 Change Of Break Location

A run was made where the break location was altered. The result of this run is shown in Figures 5.10, and 5.11. It was found that the results of the RELAP5 calculation are very much dependent on the location of the break. When the break location in the vessel is lowered, a larger amount of water flows to the containment in the first part of the transient. As is shown in Chapter 4, 'SCOPING ANALYSES', the level at the end of the transient depends on the amount of water leaking out of the system. The break location is in fact the largest uncertainty in this analysis.

5.6.3 5" Break

The total area of all the thin pipes equals to a 5" break, which is considered to be a medium LOCA. The only difference here is the very thin piping of the break flow geometry, or the ECCS. To examine the impact of the small hydraulic diameters on the mass flow rate, and pressure, of the very thin piping, a calculation was made in which the break flow piping was treated as one big pipe with a diameter of 5".

In Figures 5.12 and 5.13 the pressure and the break mass flow rate, compared with the base case. Because of the higher mass flow rate, the pressure in the reactor vessel decreases at a faster rate than in the base case.

5.6.4 Junction 319

Another feature of the model that was examined was the junction that connects the top of the fuel elements to the top of the moderator (CCC=319), and it was found to have an equalizing effect on the pressure (and water level) of the fuel elements and the moderator. When junction 319 is not present, the mass flow rates at the break results in a lower void fraction (a higher water level) than in the base case. The larger amount of water leads to a faster progression of the initial part of the transient, and, therefore, the steam formation in the steam generator starts at an earlier point, around 900 s, instead of 1000 s in the base case. See Figures 5.14 and 5.15.

5.6.5 Reactor Coolant Pump Not Tripped

A calculation was made without the tripping the pump, in order to see its effect on the break mass flow rate and the pressure in the system. The pump keeps the water level in the fuel elements high, and, therefore, water is leaking out for a longer period than in the base case.

This means that the pressure, and the mass flow rate is kept at a higher level. If the pumps would have been kept on during the incident, the water level in the reactor vessel would become lower (compared to the base case) after the equilibrium is reached between the containment and reactor vessel.

In any case, the pump could not be operated during the whole transient, because of the voiding, and subsequent cavitation. Furthermore, loss of off-site power is assumed for this hypothetical scenario, and there were not enough power generated by the diesel generators to run the main coolant circulation pumps.

5.7 Discussion

The RELAP5/MOD3.1 code has wide range of validation with experiments, and is generally a reliable code. In the present application, however, with a complicated break-flow geometry having very thin pipes with an hydraulic diameters between 1.5 – 51 mm, not much validation work has been reported. The question whether two phase critical flow in small pipes having large pressure drop at high two-phase flow velocities is well-predicted by RELAP5, might have an impact on the results. Validation of the critical flow model used can be found in the manual of RELAP5/MOD3.1. The diameters in those experiments are not as small as in this particular model.

The modeling of the location of the break flow geometry inlet in the RELAP5/MOD3.1 model of Ågesta is a source of uncertainty, which has a considerable impact on the final collapsed water level in the reactor vessel. Some analyses of the impact of the break location has been made in this report and it indicates varying break mass flow rates for different break locations. The junction, which connects the top of the moderator with the top of the fuel channel has an impact on the mass flow rate of the break as well, because of its equalizing effect on the pressure and water level for the moderator and for the coolant in the fuel channel.

The base case may not be the most realistic course of events in this incident. At the time of the incident, the plant was flooded with water, and the control system was not in perfect functioning mode. The time duration between the hypothetical break and the SCRAM are assumed to be short.

After the flow ceases to be critical, break flow can be calculated by the TEENAGE model described in Chapter 4 of this report. This calculation has been performed in order to estimate the progression of the transient after 6000 s. Of main interest is the time elapsed until pressure equilibrium is reached between the reactor vessel and the containment, and the amount of water that has escaped into the containment by that time. The collapsed level in the vessel after the blowdown can then be calculated.

Table 5.1: Heavy and Light Water Properties.

Temperature=100 (°C)								
	p_{sat} (bar)	ρ' (kg/m^3)	ρ'' (kg/m^3)	C_p (kJ/kg)	h' (kJ/kg)	h'' (kJ/kg)	k (W/mK)	ν (10^6m^2/s)
H_2O	1.014	957.9	0.6188	4.216	419.1	2674.5	0.681	0.295
D_2O	0.9649	1063.5	0.6321	4.182	401.9	2472.4	0.643	0.309
Temperature=200 (°C)								
	p_{sat} (bar)	ρ' (kg/m^3)	ρ'' (kg/m^3)	C_p (kJ/kg)	h' (kJ/kg)	h'' (kJ/kg)	k (W/mK)	ν (10^6m^2/s)
H_2O	15.54	865.1	8.019	4.497	852.3	2792.3	0.666	0.158
D_2O	15.46	957.9	8.673	4.326	826.4	2584.5	0.597	0.161
Temperature=240 (°C)								
	p_{sat} (bar)	ρ' (kg/m^3)	ρ'' (kg/m^3)	C_p (kJ/kg)	h' (kJ/kg)	h'' (kJ/kg)	k (W/mK)	ν (10^6m^2/s)
H_2O	33.45	813.7	16.74	4.760	1037.3	2803.6	0.636	0.142
D_2O	33.61	900.4	18.73	4.555	1007.4	2598.3	0.554	0.146

Table 5.2: Primary system condition in steady state.

Mass flow rates (kg/s)		
Part of the plant	According to [1]	Steady state run
Main Pump	1200.0	1200.0
Fuel element	996.0	995.0
Junction 315 (see Figure 5.1)	194.2	204.6
Velocities (m/s)		
Part of the plant	According to [1]	Steady state run
Main Piping (225)	7.0	8.1
Main Piping (275)	4.6	5.3
SG-tubes	3.2	2.2
Moderator	0.08	0.08
Central Element	2.7	2.7
Pressures (bar)		
Part of the plant	According to [1]	Steady state run
Bottom, inlet	36.1	35.7
Water Spreader (in)	35.5	35.7
Fuel Elements (out)	34.0	34.1
Moderator (top)	34.0	34.1
Moderator (bottom)	34.8	34.4
Bottom, outlet	34.5	34.4
Pump, inlet	32.3	32.4
Pump, outlet	35.5	36.2

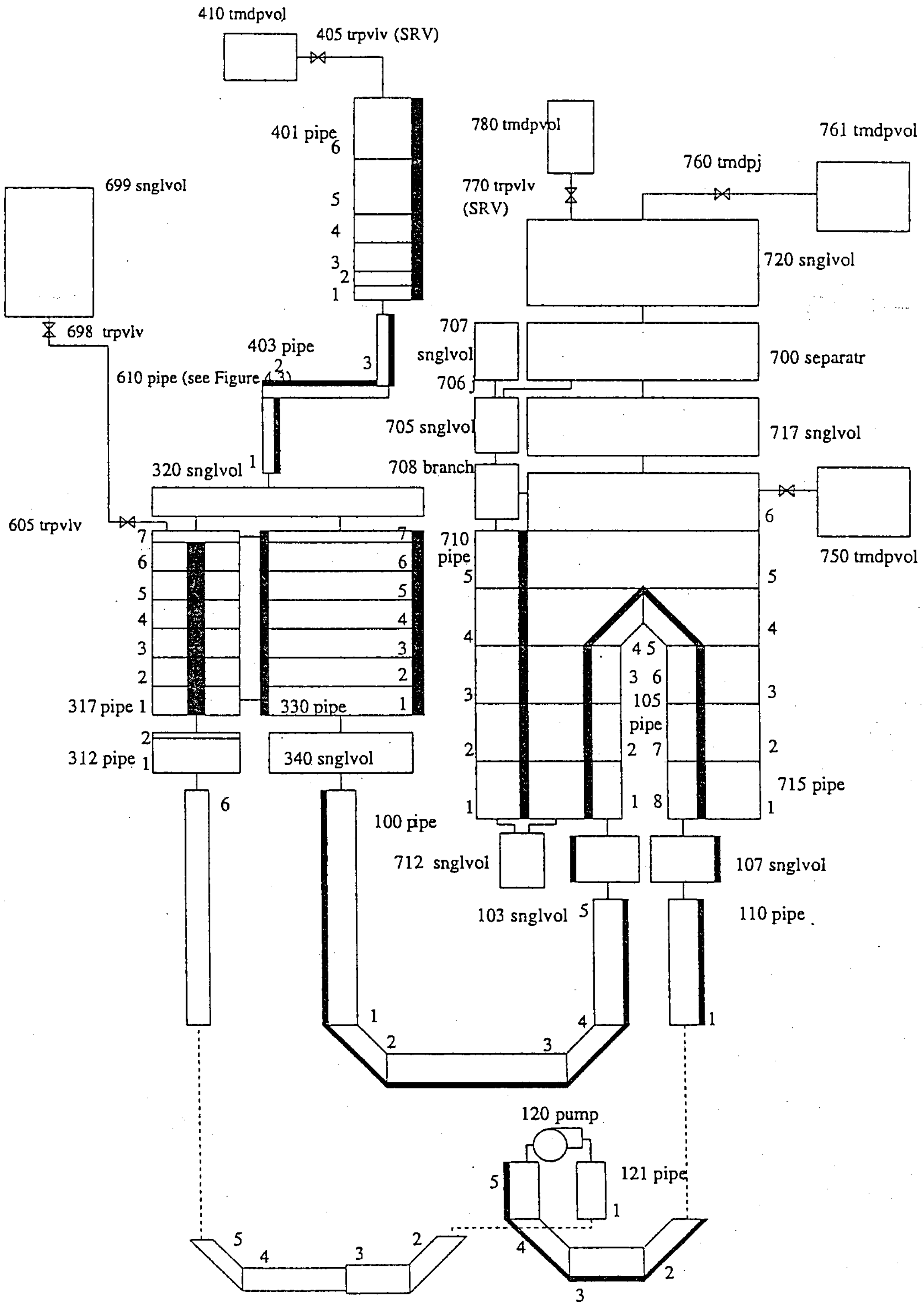


Figure 5.1: Nodalization of Ågesta for RELAP5/MOD3.

Table 5.3: Steady State Conditions of the Primary and Secondary Side.

Reactor Power (MW): 65.0			
Pressures (bar)			
Primary side (Pressurizer)		*Secondary side*	
34.0		15.5	
Total Volume (m^3)			
Primary side		*Secondary side*	
86.0		87.8	
Total Mass (kg)			
Primary side		*Secondary side*	
63,200		33,100	
Mass flow rates (kg/s)			
Primary side		*Secondary side*	
Part of the plant	Steady state	Part of the plant	Steady state
Main Pump	1025	Feed water in	30.1
Fuel element	858	Steam out	30.1
Junction 315	167	Boiling region (Bottom)	290
		Downcomer	290
Velocities (m/s)			
Primary side		*Secondary side*	
Part of the plant	Steady state	Part of the plant	Steady state
Main Piping (275)	4.61	Feed water in	1.56
Main Piping (225)	6.75	Steam out	3.27
SG-tubes	2.55	Boiling region (Bottom)	0.056/0.78
Central Element	2.37	Boiling region (Top)	0.055/1.32
Moderator	0.07	Steam dome	0.40
Temperatures (K)			
Primary side		*Secondary side*	
Part of the plant	Steady state	Part of the plant	Steady state
Bottom, inlet	476.8	Feed water in	413.1
Fuel Elements (out)	493.7	Downcomer (top)	472.9
Moderator	493.7	Boiling region (bottom)	474.3
Inlet SG	491.4	Separator	473.3
Outlet SG	476.9	Steam dome	473.2
Cold Leg	476.9	Steam out	473.2
Pressures (bar)			
Primary side		*Secondary side*	
Part of the plant	Steady state	Part of the plant	Steady state
Bottom, inlet	35.5	Feed water in	15.5
Water Spreader	35.1	Downcomer (top)	15.6
Fuel Elements (out)	34.4	Boiling region (bottom)	15.8
Moderator (top)	34.4	Separator	15.5
Moderator (bottom)	34.7	Steam dome	15.5
Bottom, outlet	34.7	Steam out	15.5
Pump, inlet	33.0		
Pump, outlet	35.8		

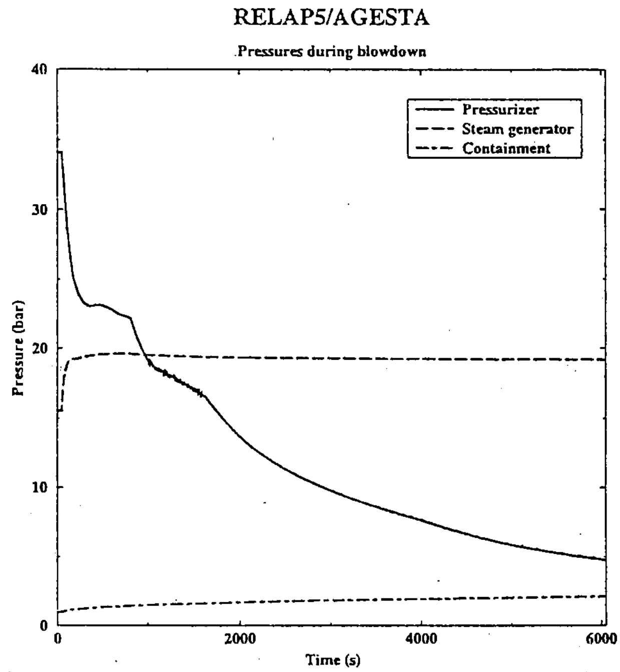


Figure 5.2: RELAP5: Pressure during the blowdown.

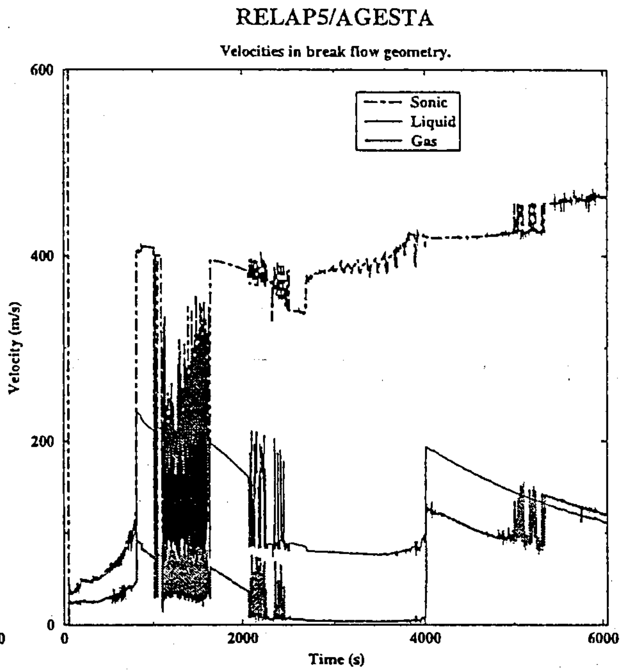


Figure 5.4: RELAP5: Velocities in the break flow geometry.

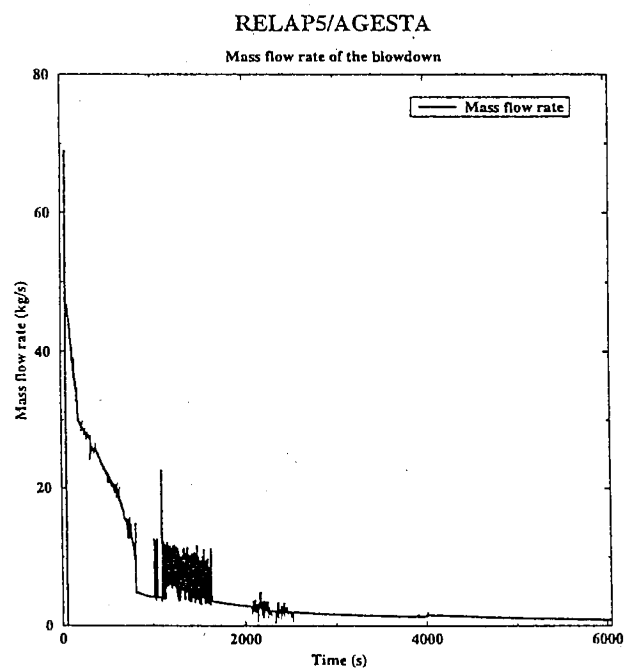


Figure 5.3: RELAP5: Mass flow rate of the blowdown.

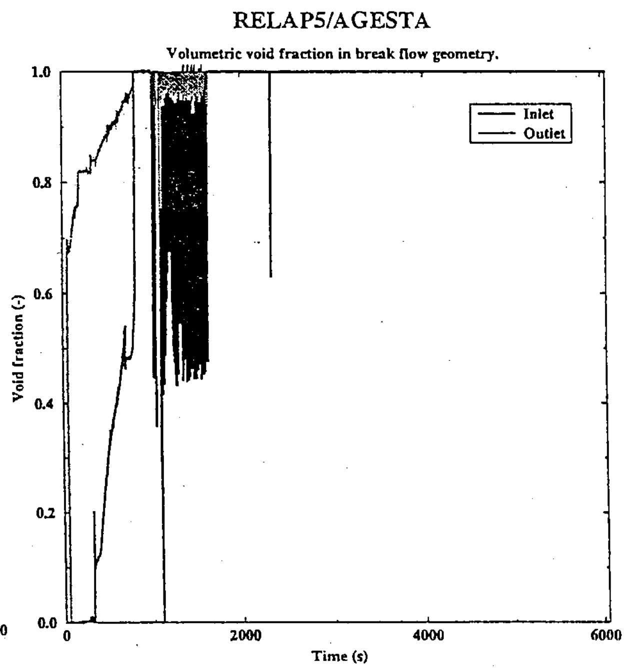


Figure 5.5: RELAP5: Void fraction in break flow geometry.

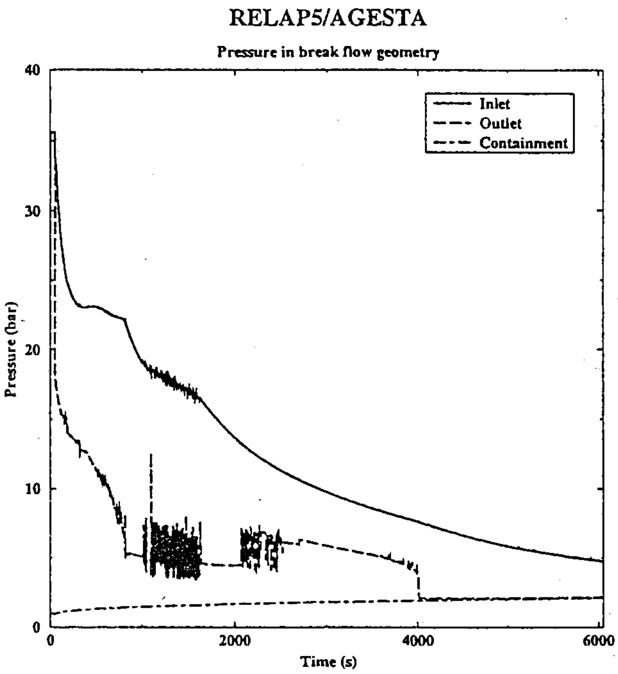


Figure 5.6: RELAP5: Pressure in break flow geometry.

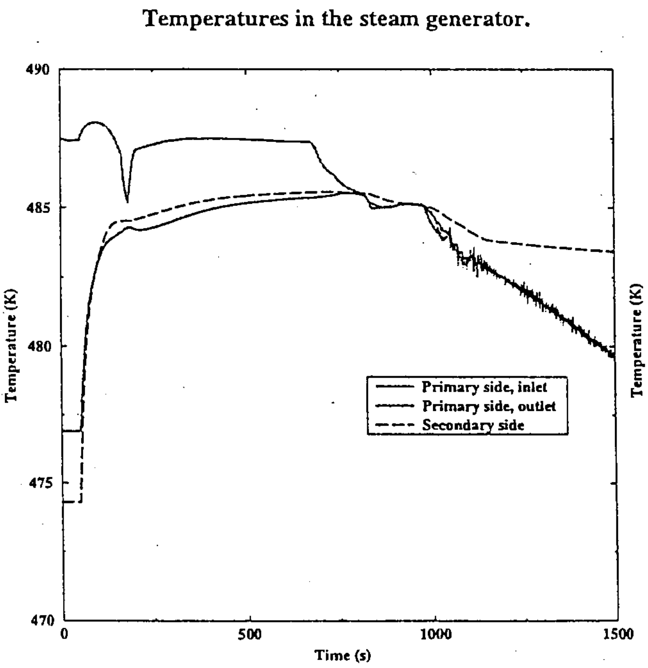


Figure 5.7: RELAP5: Primary and Secondary Steam Generator Temperatures

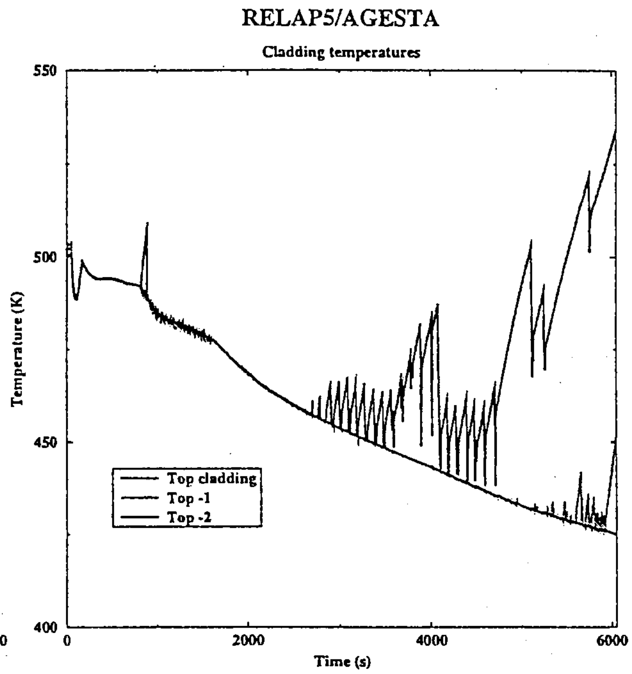


Figure 5.8: RELAP5: Cladding Temperatures

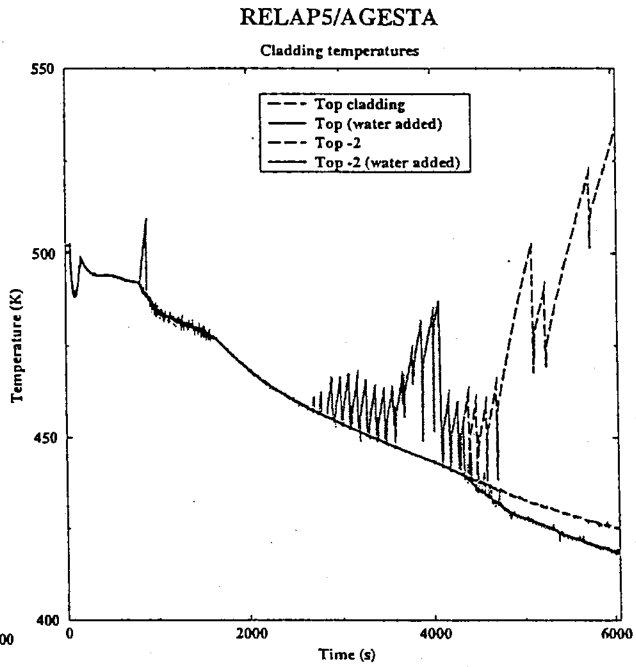


Figure 5.9: Cladding temperature when water is added.

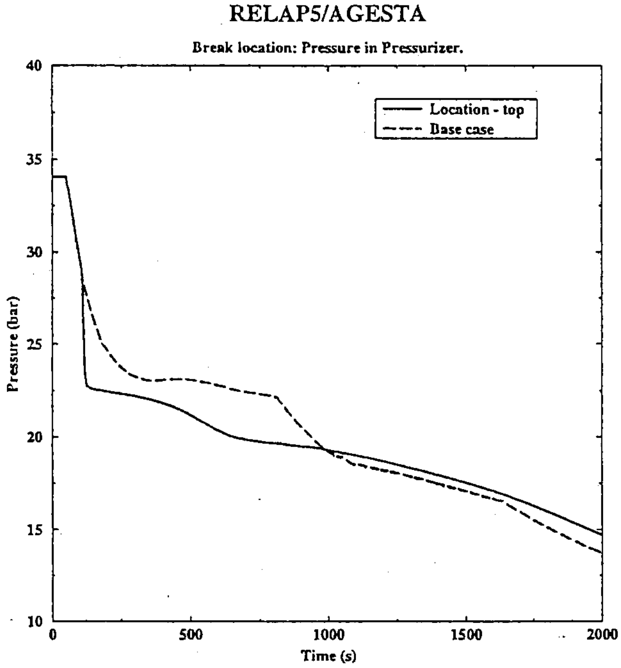


Figure 5.10: RELAP5: Comparison of pressure with different break locations.

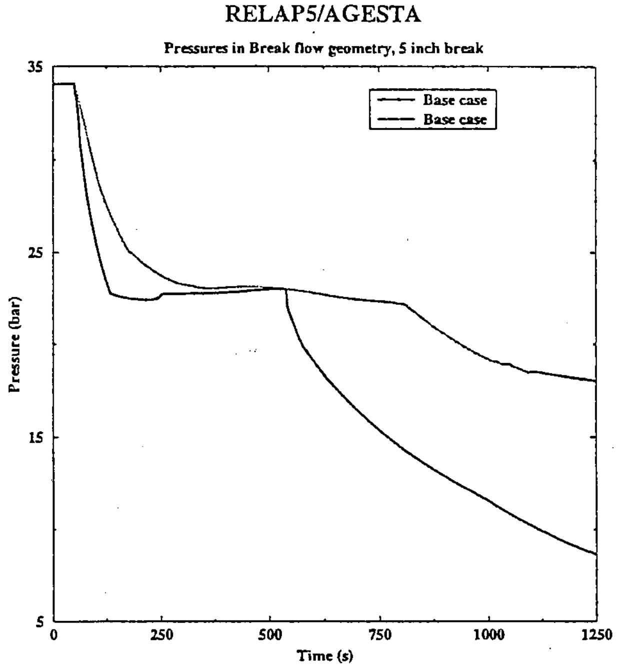


Figure 5.12: RELAP5: Pressure of 5" break and base case

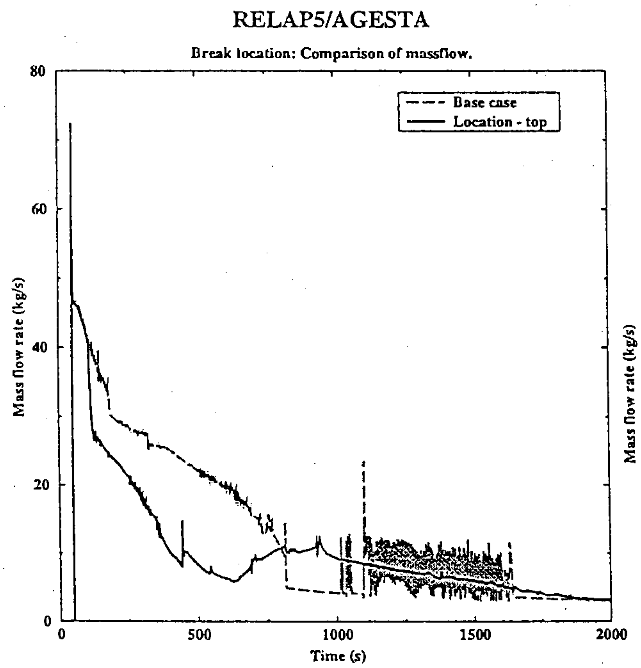


Figure 5.11: RELAP5: Mass flow rates with different break locations.

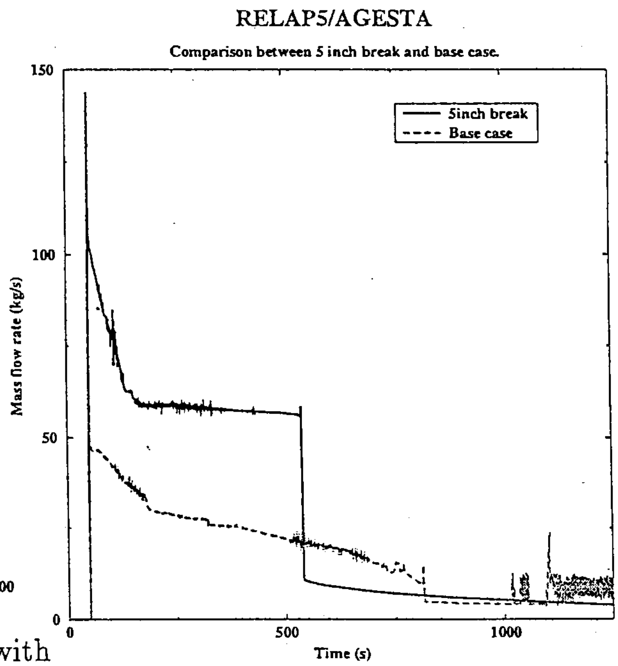


Figure 5.13: RELAP5: Mass flow rates of 5" break and base case

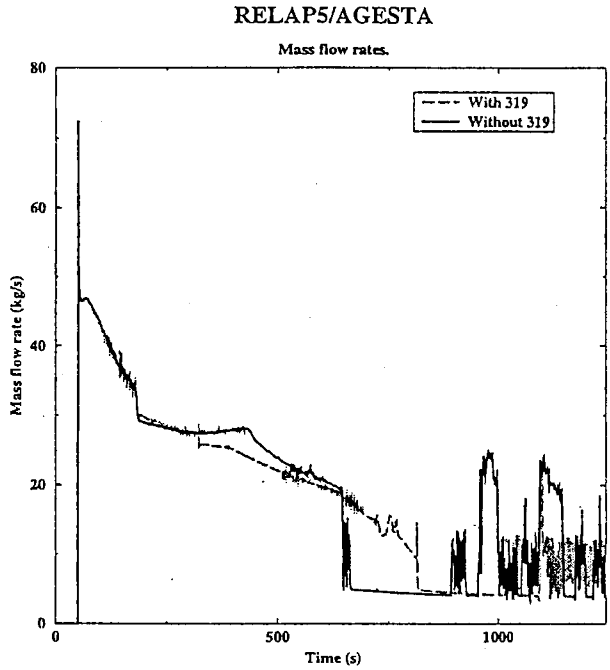


Figure 5.14: RELAP5: Mass flow rates with and without junction 319

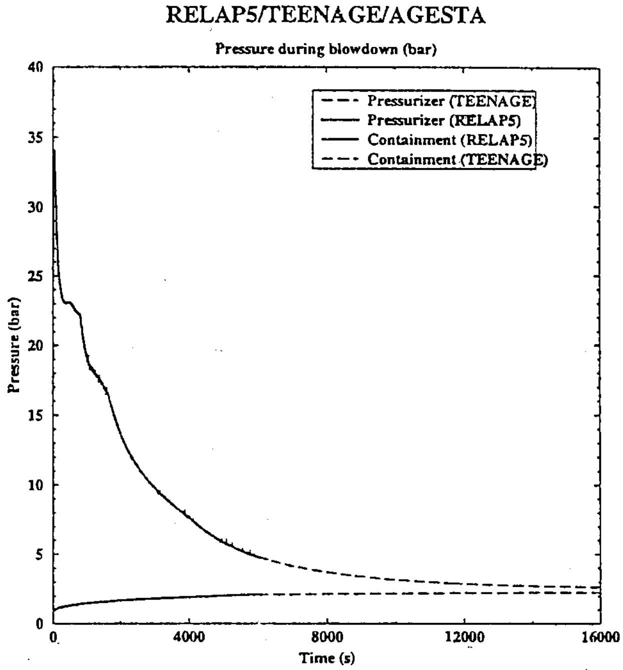


Figure 5.16: Extension of RELAP5: Pressures

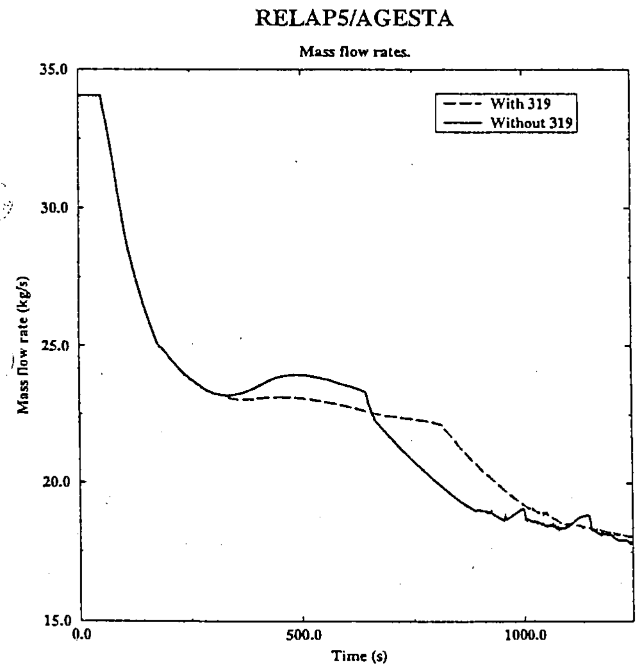


Figure 5.15: RELAP5: Pressures with and without junction 319

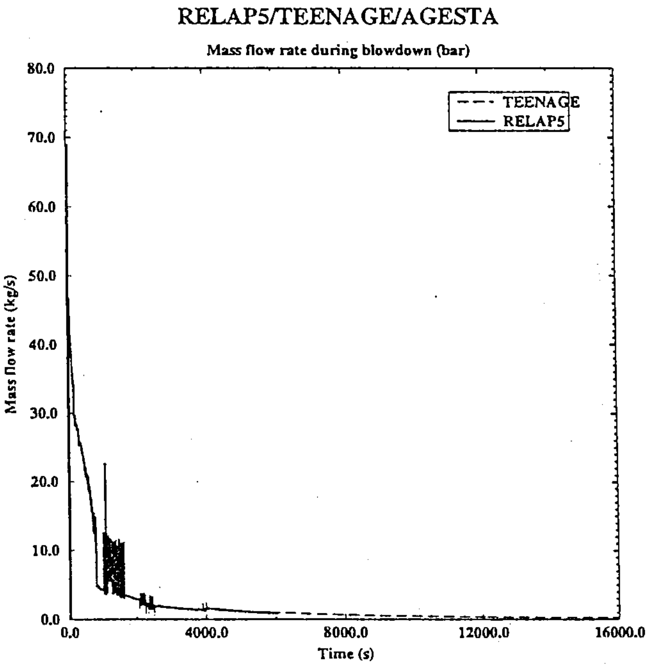


Figure 5.17: Extension of RELAP5: Break mass flow rates

Chapter 6

SUMMARY

In order to get an overall picture of the hypothetical accident, the different parts of the analysis can be combined. The RELAP5/MOD3.1 calculation will serve as the best estimate calculation for the initial phase of the transient, when two phase critical flow prevalent, and heat transfer from secondary side to primary side occurs in the steam generator. When the break flow no longer is restricted by critical flow, the model in Chapter 4 (TEENAGE) can be used. This model is used in order to estimate the time elapsed, and the level in the reactor vessel, until pressure equilibrium is reached between the reactor vessel and the containment. After this, coolant boil-off takes place, which is a very slow process; especially in this case, where the volume ratio of the moderator and fuel is very large, and the power of the core is relatively low.

After the first 6000 seconds of the transient, the normalized collapsed level of the water in the reactor tank is 0.76 (where 1.0 denotes the top of the core, and 0.0 denotes the bottom). After 16,000 seconds, when the pressure in the containment and the vessel are equal, the level is 0.73. The level is between 0.44 – 0.52, 24 hours later, respectively, for the decay heat levels of 1.0% and 0.67%. The time to reach a normalized level of 0.1 is 87.2 hours (3.6 days) and 128 hours (5.3 days) depending on decay heat level.

We believe that the system for filling and emptying the primary side of its water at low pressure could be functioning with an external light water source. Therefore, it is highly unlikely that Zircaloy oxidation, and subsequent temperature increase would take place in this incident.

Figures 6.1 and 6.2, show the collapsed level as a function of time after the break.

PRIMARY SYSTEM LEVELS

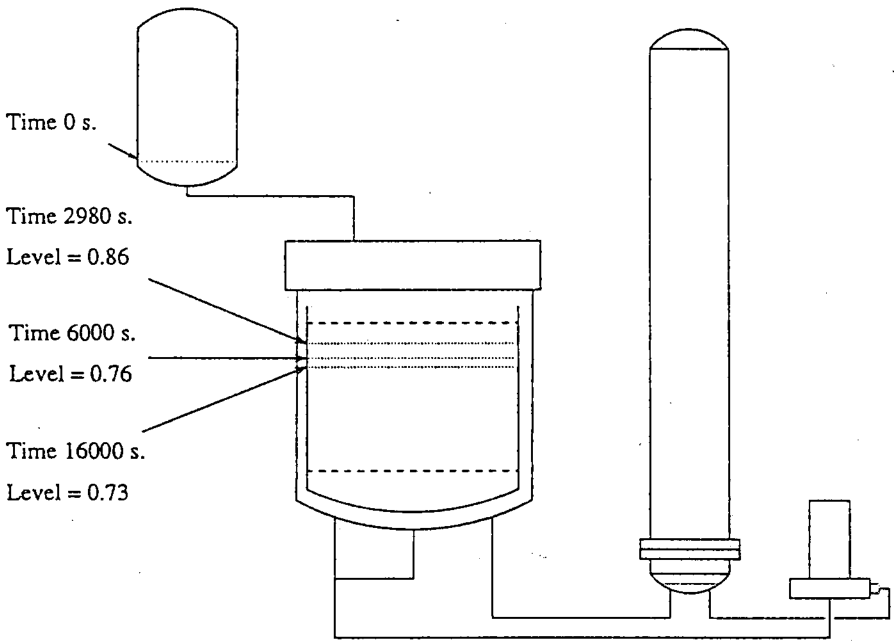


Figure 6.1: Collapsed level in Ågesta before boil-off.

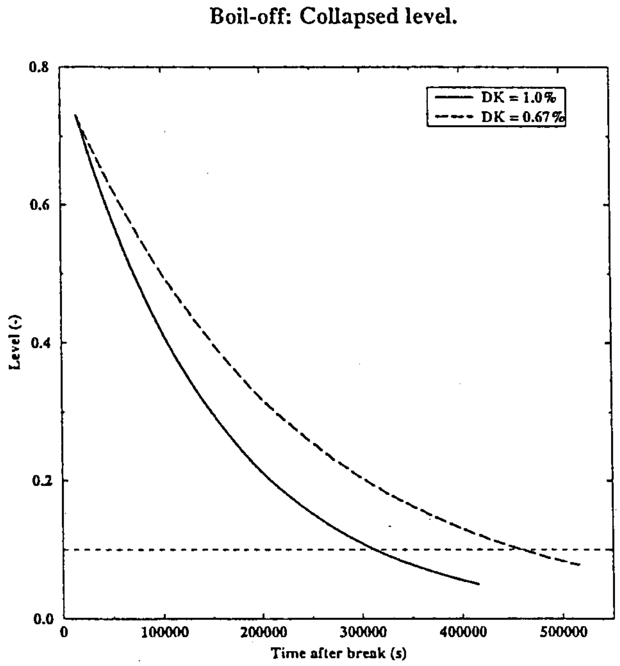


Figure 6.2: Boil-off: Collapsed level.

Bibliography

[1] B. Lilliehöök; *Ågesta Kraftvärmeverk, Sammanställning av Tekniska Data och Beskrivningar mm För Reaktordelen*; AB Atomenergi; Stockholm; 1964. (In Swedish)

[2] B. McHugh et al.; *The Ågesta Nuclear Power Station, a Staff Report by AB Atomenergi*; AB Atomenergi; Stockholm; 1964.

[3] K. Karlsson; *Värmetekniska Beräkningar för R3/ADAMs Katastrofkylanläggning*; PM A-49/59; Statens Vattenfallsverk; Stockholm; 21 Oct 1959. (In Swedish)

[4] W. Hasling; *Ågesta Kraftvärmeverk Nödkylsystem P214 Funktionsanalys*; PM V-79/62; Staten Vattenfallsverk; Stockholm; 28 May 1962. (In Swedish)

[5] L. O. Wredberg; *R3/Adam — Katastrofkylanläggning*; PM A-B5/59; Statens Vattenfallsverk; Stockholm; 2 Oct 1959. (In Swedish)

[6] H. Höglund; *Förslag Till Ombyggnad av Sprinklersystemet för Reaktortank*; Arbetsrapport 108/1968 (reg no. 8242); Vattenfall; Stockholm; 14 June 1968. (In Swedish)

[7] E. Ericsson et al.; *Ågesta Kraftvärmeverk — Haveri på kylsystem P210 den 1/5 1969*; Arbetsrapport 156/1969; Vattenfall; Stockholm;12 May 1969. (In Swedish)

[8] K. E. Sandstedt et al.; *Ågesta kraftvärmeverk — Felfunktioner och skador på elutrustning i samband med haveri på kylsystem P210 den 1/5 1969*; Arbetsrapport 158/1969; Vattenfall; Stockholm; 30 May 1969. (In Swedish)

[9] AB Atomenergi; *Beskrivning av Reaktorstommen för Ågesta Kraftvärmeverk*; Arbetsrapport R3-198; AB Atomenergi; Stockholm; 1963. (In Swedish)

[10] K Persson et al.; *Ågesta kraftvärmeverk — Årsrapport för tiden 1 januari - 31 december 1969*; Arbetsrapport 207/1970 Vattenfall; Stockholm; 1970. (In Swedish)

[11] P. Österman; *Svår Reaktorolycka i Ågesta*; Dagens Nyheter, 13th April 1993. (In Swedish)

[12] T. Okkonen; *A Synthesis Of Hydrogen Behaviour In Severe Reactor Accidents*; Report STUK-YTO-TR-59; STUK (Finnish Center For Radiation and Nuclear Safety); 1993; Helsinki, Finland.

[13] D. F. Fletcher, B. D. Turland, and S. P. A. Lawrence; *A Review of Hydrogen Productions During Melt/Water Interaction in LWRs*; Nuclear Safety, Vol 33, No 4, pp 514-534; 1992;

[14] W.J. Garland and B.J. Hand; *Simple Functions for the Fast Approximation of Light Water Thermodynamic Properties*; Nuclear Engineering and Design 113, 1989 (pp 21–34).

[15] C. D. Fletcher, R.R. Schultz; *RELAP5/MOD3 User's Guidelines*; ISBN 0-16-036098-6; 1992; Washington D.C., USA

[16] I. E. Idelchick; *Handbook of Hydraulic Resistance*, 2nd ed; Hemishpere Pub. Corp.; 1986; Washington, D.C., USA

[17] Ya. Z. Kazavchinskii; *Heavy Water Thermophysical Properties*; Israel Program for Scientific Translations; 1971; Jerusalem, Israel.

[18] L. Wester; *Tabeller och Diagram för Energitekniska Beräkningar*; 1990; Västerås, Sweden.

Appendix A: RELAP5/MOD3.1 Input for Ågesta Nuclear Power Plant

```
=Steady state tryout for different subsystems
*****************************************************************
*****    RELAP5 INPUT FOR AAGESTA NUCLEAR POWER PLANT      ******
*****************************************************************
*    Problem Type Option
100      new         transnt
*100      restart      transnt
*             Units  Input/Output
102      si       si
* Restart Input File Control card
*      Restart number
*103     1574312
* Restart-plot file Control card
*Action
*104 none
* Noncondensible gas type (Air ?)
110      air
* Hydrodynamic Control Cards
*              Ref.Vol#      Ref.Elev      Fluid Type     Name Of Sys.
120      100010000      0.0             d2o              PrImAr
121      712010000      0.0             d2o              SeCoNd
*
*****************************************************************
* Time Step Control
*              End Time Min.Tmstp Max.Tmstp Contr. Edit Freq. Restart
203      50.0     1.0e-7     0.0500      15000    100    500      500
204      55.0     1.0e-7     0.0050      15000    10    5000     5000
205     550.0     1.0e-7     0.0050      15000    100    5000     5000
206    2300.0     1.0e-7     0.0010      15000    1000  15000    15000
207    6050.0     1.0e-7     0.0010      15000    5000  60000    60000
*
*****************************************************************
*       CONTROL VARiABLES
*****************************************************************
*
* Control Variable Card Type
*              999/9999
20500000  9999
*
*  3171 summerar heatflux*area for fuel elements
20531710  heattr    sum  1.0     0.0     0
20531711  0.0     56.163  htrnr   317100101
20531712          56.163  htrnr   317100201
20531713          56.163  htrnr   317100301
20531714          56.163  htrnr   317100401
20531715          56.163  htrnr   317100501
20531716          56.163  htrnr   317100601
*
*  1050 summerar heatflux*area for s.g tubes, left side
20510500  heattr    sum  1.0     0.0     0
20510501  0.0      205.47 htrnr     105100100
20510502           205.47  htrnr    105100200
20510503           205.47  htrnr    105100300
20510504           205.47  htrnr    105100400
20510505           205.47  htrnr    105100500
20510506           205.47  htrnr    105100600
20510507           205.47  htrnr    105100700
20510508           205.47  htrnr    105100800
```



```
*
*   1051 summerar heatflux*area for s.g tubes, right side
20510510  heattr    sum  1.0     0.0     0
20510511  0.0     255.58  htrnr   105100101
20510512          255.58  htrnr   105100201
20510513          255.58  htrnr   105100301
20510514          255.58  htrnr   105100401
20510515          255.58  htrnr   105100501
20510516          255.58  htrnr   105100601
20510517          255.58  htrnr   105100701
20510518          255.58  htrnr   105100801
*
*****************************************************************
*       6101  Integrates the massflow in junction 610010000
*****************************************************************
*
* Control Component Type Card
*         Name     Type       Scale Initial  I.v.flag    Limiter  min max
20561010 Mass     integral   1.0   0.0       0           0
* Control Component Data Cards
*         Alphanumeric   Integer
20561011 mflowj          610010000
*
*****************************************************************
*       6117  Integrates the massflow in junction 610170000
*****************************************************************
*
* Control Component Type Card
*         Name     Type       Scale Initial  I.v.flag    Limiter  min max
20561170 Mass     integral   1.0   0.0       0           0
* Control Component Data Cards
*         Alphanumeric   Integer
20561171 mflowj          610170000
*
*****************************************************************
*       6980  Integrates the massflow in junction 698000000
*****************************************************************
*
* Control Component Type Card
*         Name     Type       Scale Initial  I.v.flag    Limiter  min max
20569800 Mass     integral   1.0   0.0       0           0
* Control Component Data Cards
*         Alphanumeric   Integer
20569801 mflowj          698000000
*
*****************************************************************
*               Minor Edit Variables
*****************************************************************
*
301         cntrlvar   1050
302         cntrlvar   1051
303         cntrlvar   3171
*
*****************************************************************
*       Trips
*****************************************************************
*     502 - Break trip, ECCS breaks after this time.
*
```

```
502     time   0  gt  null   0    50.0   1   -1.0
602     -502   and   -502    n
*
*****************************************************************************
*     503 - Manual SCRAM and pump trip at t=300s
*
503     time   0  gt  null   0    50.0   1   -1.0
603    -503   and   -503    n
*
*****************************************************************************
*     505, 506, 605-607 press in Pressurizer > 42.0 bar
*
505     p   401060000  gt  null    0   41.0e+5   1   -1.0
506     p   401060000  gt  null    0   42.0e+5   1   -1.0
605     -607   and   506   n
606      607   and   505   n
607      605   or    606   n
*
*****************************************************************************
*     510, 511, 610-612 - pressure in the steam generator.
*
510     p 720010000    gt  null   0   28.0e+05   n   -1.0
511     p 720010000    gt  null   0   30.0e+05   n   -1.0
610     -612   and   511   n
611      612   and   510   n
612      610   or    611   n
*
*****************************************************************************
*            Hot leg of the primary side.
*****************************************************************************
*  OMPONENT NAME    COMPONENT TYPE
*
1000000 hotleg        pipe
* Pipe Information  Card
*             Number Of Volumes
1000001 5
* Pipe Volume Flow Areas
*             Volume Flow AreasVol. num
1000101 0.2376                 5
*             Junction Flow AreasJunc. Num
1000201 0.2376                 4
* Pipe Volume Flow Lengths
*             Volume Flow LengthsVol. num
1000301 4.2                    1
1000302 1.4142                 2
1000303 3.2                    3
1000304 1.4142                 4
1000305 2.2                    5
* Pipe Volume Flow AZIMUTHAL Angles
*             Volume Flow AngleVol. Num
1000501 0.0                    5
* Pipe Volume Flow Inclination Angles
*             Volume Flow AngleVol. Num
1000601 -90.0                  1
1000602 -45.0                  2
1000603   0.0                  3
1000604  45.0                  4
1000605  90.0                  5
```



```
* Pipe Volume Flow Friction Data
*             Volume Roughn.Hydr. Diam.    Vol. Num
1000801 4.57e-5         0.275             5
* Pipe Volume Flow Energy Loss Coefficient
*             Forward Loss  Reverse Loss  Junc. Num
1000901 0.0             0.0               4
* Pipe Volume Flow Control Flags
*             pvbfe         Vol. Num
1001001 00000       5
* Pipe Junction Flow Control Flags
*             fvcahs        Junc. Num
1001101 000000       4
* Pipe Volume Flow Initial Conditions
*             ebt            Pres.          Temp.   Vol. Num
* p2nd=15.5bar
1001201 200  3.48208e+06    9.01789e+05  2.41943e+06  0.00  0.0       1
1001202 200  3.50454e+06    9.01790e+05  2.41941e+06  0.00  0.0       2
1001203 200  3.50800e+06    9.01792e+05  2.41941e+06  0.00  0.0       3
1001204 200  3.50230e+06    9.01792e+05  2.41941e+06  0.00  0.0       4
1001205 200  3.48677e+06    9.01794e+05  2.41942e+06  0.00  0.0       5
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
1001300 1
* Pipe Junction Initial Conditions
*             Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
* p2nd=15.5bar
1001301 1025.1          0.0           0.0             4
*
*****************************************************************************
*      Junction from hotleg to the inlet plenum of the steam generator.
*****************************************************************************
*     COMPONENT NAME    COMPONENT TYPE
*
1020000 hl-sg           sngljun
*             From                        To
Flow area loss coeff.cntrl var.
1020101 100010000       103000000       0.2376     0.0  0.0       0000
*             Control var.
1020110 0.0      0.0       1.0      1.0
*             cntr.         Init.mass.flow
* p2nd=15.5bar
1020201 1        1025.1     0.0       0.0
*
*****************************************************************************
*             Inlet plenum of the steam generator.
*****************************************************************************
*      COMPONENT NAME   COMPONENT TYPE
*
1030000 sgip            snglvol
*             Area Length Volume  Azi. lev. Elev(m) Roughn.   Hyd.dia
1030101 2.26195 0.8     1.80956 0.0     90.0    0.8     4.57e-6 0.010    00000
*       ebt   Pres.        Temp
1030200 200   3.48222e+06    9.01794e+05  2.41942e+06 0.000    0.0
*
*****************************************************************************
*             Junction from inlet plenum to the steam gen. primary tubes
*****************************************************************************
*             COMPONENT NAMECOMPONENT TYPE
```

```
*
1040000 sglp-tu        sngljun
*        From          To              Flow area   loss coeff.  cntrl var.
1040101 103010000     105000000        0.602532        43.0  43.0      0000
*             Control var.
1040110 0.0      0.0      1.0      1.0
*        cntr. Init.mass.flow
1040201 1       1025.1     0.0      0.0
*
*******************************************************************
*             Steam generator primary tubes
*******************************************************************
*    COMPONENT NAME     COMPONENT TYPE
*
1050000 sgtubes          pipe
* Pipe Information  Card
*             Number Of Volumes
1050001 8
* Pipe Volume Flow Areas
*             Volume Flow AreasVol. num
1050101 0.42121             8
*             Junction Flow AreasJunc. Num
*1050201 0.602532              7
1050201 0.42121             7
* Pipe Volume Flow Lengths
*             Volume Flow LengthsVol. num
1050301 1.0                   8
* Pipe Volume Flow Inclination Angles
*             Volume Flow AngleVol. Num
1050601  90.0                    3
1050602  58.5                    4
1050603 -58.5                    5
1050604 -90.0                    8
* Pipe Volume Flow Friction Data
*             Volume Roughn.Hydr. Diam.    Vol. Num
1050801 4.57e-5         0.0082              8
* Pipe Loss coefficient Data
*        Forward         Reverse          Junc. num
1050901  0.0             0.0                 7
*             pvbfe           Vol. Num
1051001 10100       8
* Pipe Junction Flow Control Flags
*             fvcahs          Junc. Num
1051101 000000    7
* Pipe Volume Flow Initial Conditions
*             ebt             pres.           Temp
* p2nd = 15.5 bar
1051201  203  3.39753e+06  487.492   0.0   0.00      0.0  1
1051202  203  3.37592e+06  484.533   0.0   0.00      0.0  2
1051203  203  3.35431e+06  482.258   0.0   0.00      0.0  3
1051204  203  3.33339e+06  480.486   0.0   0.00      0.0  4
1051205  203  3.32112e+06  479.122   0.0   0.00      0.0  5
1051206  203  3.31750e+06  478.122   0.0   0.00      0.0  6
1051207  203  3.31460e+06  477.402   0.0   0.00      0.0  7
1051208  203  3.31171e+06  476.899   0.0   0.00      0.0  8
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
1051300 1
```



```
* Pipe junction Initial Conditions
*             Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
1051301 1025.1     0.0        0.0     7
*
*******************************************************************
*             Junction from s.g. primary tube to the outlet plenum
*******************************************************************
*    COMPONENT NAME   COMPONENT TYPE
*
1060000 tu-sgopl         sngljun
*        From           To              Flow area  loss coeff.  cntrl var.
1060101 105010000      107000000        0.60253      14.00    14.00     0000
*
1060110 0.0      0.0       1.0      1.0
*             cntr.        Init.mass.flow
1060201 1       1025.1     0.0      0.0
*
*******************************************************************
*             Outlet plenum
*******************************************************************
*    COMPONENT NAME   COMPONENT TYPE
*
1070000 sgopl    snglvol
*       Area    Length Volume  Azi.     Elev.     Elev(m) Roughn. Hyd.dia
1070101 2.26195 0.8     0.0     0.0      -90.0     -0.8    4.57e-6  0.010   00000
*             ebt      Pres         Temp
1070200 200      3.29555e+06 8.38648e+05   2.41956e+06  0.00
*
*******************************************************************
*             Junction from the outlet plenum to cold leg
*******************************************************************
*    COMPONENT NAME   COMPONENT TYPE
*
1080000 sgopl-cl         sngljun
*             From          To               Flow area  loss coeff.  cntrl var.
1080101 107010000      110000000        0.2376    0.0       0.0       0000
*
1080110 0.0      0.0       1.0      1.0
*             cntr.        Init.mass.flow
1080201 1       1025.1     0.0      0.0
*
*******************************************************************
*             Cold leg, pump suction side.
*******************************************************************
*    COMPONENT NAME COMPONENT TYPE
*
1100000 coldleg          pipe
* Pipe Information  Card
*             Number Of Volumes
1100001 5
* Pipe Volume Flow Areas
*             Volume Flow AreasVol. num
1100101 0.2376             5
*             Junction Flow AreasJunc. Num
1100201 0.2376             4
* Pipe Volume Flow Lengths
*             Volume Flow LengthsVol. num
1100301 2.2                1
```

```
1100302 1.4142          2
1100303 1.3             3
1100304 1.4142          4
1100305 1.0             5
* Pipe Volume Flow AZIMUTHAL Angles
*             Volume Flow AngleVol. Num
1100501 0.0             5
* Pipe Volume Flow Inclination Angles
*             Volume Flow AngleVol. Num
1100601 -90.0           1
1100602 -45.0           2
1100603   0.0           3
1100604  45.0           4
1100605  90.0           5
* Pipe Volume Flow Friction Data
*             Volume Roughn.Hydr. Diam.    Vol. Num
1100801 4.57e-5         0.275          5
* Pipe Volume Flow Energy Loss Coefficient
*             Forward Loss  Reverse Loss  Junc. Num
1100901 0.0             0.0            4
* Pipe Volume Flow Control Flags
*             pvbfe          Vol. Num
1101001 0        5
* Pipe Junction Flow Control Flags
*             fvcahs         Junc. Num
1101101 0        4
* Pipe Volume Flow Initial Conditions
*             ebt            Pres.          Temp                   Vol. Num
* p2nd = 15.5 bar
1101201 200  3.29930e+06  8.38649e+05  2.41956e+06     0.0     0.0    1
1101202 200  3.31342e+06  8.38650e+05  2.41956e+06     0.0     0.0    2
1101203 200  3.31745e+06  8.38651e+05  2.41956e+06     0.0     0.0    3
1101204 200  3.31212e+06  8.38651e+05  2.41956e+06     0.0     0.0    4
1101205 200  3.30218e+06  8.38652e+05  2.41956e+06     0.0     0.0    5
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
1101300 1
* Pipe Junction Initial Conditions
*             Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
1101301 1025.1          0.0            0.0            4
*
*****************************************************************
*      Main Coolant Pumps (all values calculated from PWR.Input.deck)
*****************************************************************
*     COMPONENT NAME    COMPONENT TYPE
*
1200000 mpump      pump
*             Pump information card
*             Flow Area Length Volume Azim.Incln.Elev.   pvbfe
1200101 0.2376    0.0    0.200  0.0     0.0     0.0     00000
*             Pump Inlet (Suction) Junction Card
*             Inlet vol.    Junc. Area    Loss Coeff.    fvhacs
1200108 110010000       0.2376         0.0     0.0    000000
*             Pump Outlet (Dicharge) Junction Card
*             Outlet vol.   Junc. Area    Loss Coeff.    fvhacs
1200109 121000000       0.2376         0.0     0.0    000000
*             Pump Volume Initial Conditions
*             ebt           Pressure       Temperature
```



```
1200200 200      3.43877e+06    8.38429e+05  2.41946e+06     0.00     0.0
*              Pump Inlet Junction Initial Conditions
*              cntrl         liq. mass flowvap. mass flowInterface velocity
(=0.0)
1200201 1       1025.1          0.0             0.0
*              Pump Outlet Junction Initial Conditions
*    cntrl  liq. mass flow  vap. mass flow Interface velocity (=0.0)
1200202 1       1025.1         0.0             0.0
*              Pump Index and Option Card
*     Data   2-phase 2-phase table torque table  pump vel. Trip  Reverse
1200301  -2     -1        -3            -1              -1      503      0
*              Pump Description Card
*              Pump vel   Init.vel/rat.vel  Rated Flow   Rat.Head
*1200302 469.7      1.0057              2040.0        10.0
*   Mflow = 1200.0 kg/s:
*1200302 500.0      1.0057              2040.0        23.6
*   Mflow = 1020.0 kg/s:  H = 0.7225*23.6
1200302 500.0      1.0057              2040.0       17.28
*              Rat.Torq  Moment of Inertia  Rated.Dens       Pump Motor Torque
*   Mflow = 1200.0 kg/s:
*1200303 3.0         1100.0             64.2              0.0
*   Mflow = 1020.0 kg/s:
1200303 1.5          6.0              40.0              0.0
*              TF2       TF0      TF1       TF3
*   Mflow = 1200.0 kg/s:
*1200304 15.0   120.0      0.0      0.0
*   Mflow = 1020.0 kg/s:
1200304 100.0   10.0     0.0     0.0
*
*
*****************************************************************
*      Coldleg.
*****************************************************************
*     COMPONENT NAME    COMPONENT TYPE
*
1210000 coldleg2         pipe
* Pipe Information  Card
*              Number Of Volumes
1210001 6
* Pipe Volume Flow Areas
*              Volume Flow AreasVol. num
1210101 0.2376          1
1210102 0.2376          4
1210103 0.1590          5
1210104 0.1590          6
*              Junction Flow AreasJunc. Num
1210201 0.2376          3
1210202 0.1590          4
1210203 0.1590          5
*Pipe Volume Flow Lengths
*              Volume Flow LengthsVol. num
1210301 1.0             1
1210302 1.4142          2
1210303 1.1             3
1210304 1.80            4
1210305 1.4142          5
1210306 4.2             6
* Pipe Volume Flow AZIMUTHAL Angles
```

```
*               Volume Flow AngleVol. Num
1210501 0.0              6
* Pipe Volume Flow Inclination Angles
*               Volume Flow AngleVol. Num
1210601 -90.0            1
1210602 -45.0            2
1210603   0.0            3
1210604   0.0            4
1210605  45.0            5
1210606  90.0            6
* Pipe Volume Flow Friction Data
*               Volume Roughn.Hydr. Diam.   Vol. Num
1210801 4.57e-5         0.275         4
1210802 4.57e-5         0.225         6
* Pipe Volume Flow Energy Loss Coefficient
*               Forward Loss  Reverse Loss  Junc. Num
1210901 0.0             0.0           5
* Pipe Volume Flow Control Flags
*               pvbfe         Vol. Num
1211001 00000         6
* Pipe Junction Flow Control Flags
*               fvcahs        Junc. Num
1211101 000000        5
* Pipe Volume Flow Initial Conditions
*               ebt           Pres.         Temp
Vol. Num
* p2nd = 15.5 bar
1211201 200  3.58473e+06  8.38429e+05    2.41935e+06    0.0     0.0     1
1211202 200  3.59352e+06  8.38430e+05    2.41935e+06    0.0     0.0     2
1211203 200  3.59760e+06  8.38431e+05    2.41934e+06    0.0     0.0     3
1211204 200  3.59691e+06  8.38432e+05    2.41934e+06    0.0     0.0     4
1211205 200  3.57884e+06  8.38434e+05    2.41936e+06    0.0     0.0     5
1211206 200  3.55075e+06  8.38439e+05    2.41938e+06    0.0     0.0     6
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
1211300 1
* Pipe Junction Initial Conditions
*               Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
1211301 1025.1           0.0             0.0            5
*
*******************************************************************
*                Inlet Volume junction (130).
*******************************************************************
*    COMPONENT NAME    COMPONENT TYPE
*
1300000  inl-jun         sngljun
* Time dependent junction geometry card
*               From                        To
Area     Loss coeff.   cntrl. var
1300101 121010000        312000000       0.15804    0.0    0.0     0000
*               cntr.         Init.mass.flow
1300201  1        1025.1     0.0      0.0
*
*******************************************************************
*        Inlet of the Vessel (312)
*******************************************************************
*   COMPONENT NAME      COMPONENT TYPE
*
```



```
3120000 inl-vess          pipe
* Pipe Information  Card
*               Number Of Volumes
3120001 2
* Pipe Volume Flow Areas
*               Volume Flow AreasVol. num
3120101 0.0              1
3120102 10.191           2
*               Junction Flow AreasJunc. Num
3120201 0.4858           1
* Pipe Volume Flow Lengths
*               Volume Flow Lengths      Vol. num
3120301 0.6              1
3120302 0.1              2
* Pipe Volume Flow Volumes
*               Volume Flow VolumesVol. num
3120401 0.4858               1
3120402 0.0                  2
* Pipe Volume Flow AZIMUTHAL Angles
*               Volume Flow AngleVol. Num
3120501 0.0              2
* Pipe Volume Flow Inclination Angles
*               Volume Flow AngleVol. Num
3120601 90.0              2
* Pipe Volume Flow Friction Data
*               Volume Roughn.Hydr. Diam.   Vol. Num
3120801 4.57e-5         0.47            1
3120802 4.57e-5         0.0             2
* Pipe Volume Flow Energy Loss Coefficient
*               Forward Loss  Reverse Loss  Junc. Num
*3120901 20.2              20.2             1
3120901 14.89             14.89            1
* Pipe Volume Flow Control Flags
*               pvbfe         Vol. Num
3121001 0         2
* Pipe Junction Flow Control Flags
*               fvcahs        Junc. Num
3121101 0         1
* Pipe Volume Flow Initial Conditions
*               ebt           Pres.          Temp    Vol. Num
* p2nd = 15.5 bar
3121201 200     3.54640e+06    8.38439e+05  2.41938e+06   0.0    0.0  1
3121202 200     3.50908e+06    8.38439e+05  2.41941e+06   0.0    0.0  2
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
3121300 1
* Pipe Junction Initial Conditions
*               Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
3121301 1025.1          0.0             0.0             1
*
*******************************************************************
*               Junction from the lower plenum to the fuel elements.
*******************************************************************
*    COMPONENT NAME     COMPONENT TYPE
*
3140000  lpl-fel          sngljun
*               From                        To
Flow area       loss coeff.   fvcahs.
```

```
3140101 312010000       317000000       0.4875        7.20  7.20  000000
*           Control var.
3140110 0.0       0.0       1.0       1.0
*           cntr.           Init.mass.flow
3140201 1        1025.1    0.0       0.0
*
***********************************************************************
*           Junction from Fuel Elements (317) to moderator (330).
***********************************************************************
*       COMPONENT NAME    COMPONENT TYPE
*
3150000 fuel-mod          sngljun
*           From              To              Flow area    loss coeff.    fvcahs.
3150101 317010003         330010004         0.27457      85.0  85.0     000003
*           Control var.
3150110 0.0      0.0        1.0        1.0
*           cntr.           Init.mass.flow
3150201 1        167.09   0.0      0.0
*
***********************************************************************
*           Fuel elements.
***********************************************************************
*    COMPONENT NAME    COMPONENT TYPE
*
3170000 fuelelem       pipe
* Pipe Information  Card
*           Number Of Volumes
3170001 7
* Pipe Volume Flow Areas
*           Volume Flow AreasVol. num
3170101 0.38783           6
3170102 0.74386           7
*           Junction Flow AreasJunc. Num
*3170201 0.38783           6
3170201 0.38783           6
* Pipe Volume Flow Lengths
*           Volume Flow LengthsVol. num
3170301 0.51              6
3170302 0.085             7
* Pipe Volume Flow Inclination Angles
*           Volume Flow AngleVol. Num
3170601 90.0              7
* Pipe Volume Flow Friction Data
*           Wall Roughn.  Hydr. Diam.    Vol. Num
3170801 4.57e-5           0.011033        1
3170802 4.57e-5           0.011033        6
3170803 4.57e-5           0.032765        7
* Pipe Volume Flow Control Flags
*           pvbfe           Vol. Num
3171001 10100        7
* Pipe Junction Flow Control Flags
*           fvcahs          Junc. Num
*3171101 000100    6
3171101 000000    6
* Pipe Volume Flow Initial Conditions
*           ebt             pres.            Temp
* p2nd = 15.5 bar
3171201 203  3.48378e+06   479.558  0.0      0.0      0.0      1
```



```
3171202 203  3.47611e+06   482.694  0.0      0.0      0.0      2
3171203 203  3.46780e+06   485.813  0.0      0.0      0.0      3
3171204 203  3.45950e+06   488.925  0.0      0.0      0.0      4
3171205 203  3.45121e+06   492.032  0.0      0.0      0.0      5
3171206 203  3.44293e+06   493.757  0.0      0.0      0.0      6
3171207 203  3.44023e+06   493.756  0.0      0.0      0.0      7
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
3171300 1
* Pipe Junction Initial Conditions
*           Init.Liq.Mass Init.Vap.Mass  Interface Vel.Junc. Num
3171301 858.04     0.0      0.0       6
*
***********************************************************************
*       Upper plenum (pipe) connected with the fuel elements (317)
*      Later: the connection to the ECCS (ccc=600) is here (branch)
***********************************************************************
*       COMPONENT NAME    COMPONENT TYPE
*
3180000 fel-upl           sngljun
*           From          To              Flow area         loss coeff.  fvcahs.
3180101 317010000         320000000         0.94114          0.0    0.0  000000
*           Control var.
3180110 0.0       0.0       1.0        1.0
*           cntr.           Init.mass.flow
3180201 1        963.40     0.0       0.0
*
***********************************************************************
*       Fuel Elements (317) connected with the Moderator (330)
***********************************************************************
*     COMPONENT NAME     COMPONENT TYPE
*
3190000 fuel-mod          sngljun
*       From              To              Flow area         loss coeff.  fvcahs.
3190101 317070003         330070004         0.48888        28.71 28.71  000000
*           Control var.
3190110 0.0       0.0       1.0        1.0
*           cntr.           Init.mass.flow
3190201 1       -105.36     0.0       0.0
*
*
***********************************************************************
*           Upper plenum
***********************************************************************
*     COMPONENT NAME      COMPONENT TYPE
*
3200000 sgopl   snglvol
*       Area   Length  Volume  Azi.   Elev. Elev(m) Roughn.   Hyd.dia
3200101 13.9205 0.45     0.0     0.0   90.0    0.45   4.57e-6  4.21    00000
*         ebt      Pres         Temp
3200200 203        3.43853e+06  493.756
*
***********************************************************************
*           Junction from the upper plenum (320) to the moderator (330).
***********************************************************************
*           COMPONENT NAMECOMPONENT TYPE
*
3210000 uppl-mod        sngljun
```

```
*       From              To                  Flow areal     oss coeff.  fvcahs.
3210101 330010000         320000000            13.007         0.0      0.0    0000
*              Control var.
*              Hydr. dia    beta      c           slope
3210110 0.0          0.0         1.0       1.0
*              cntr.         Init.mass.flow
3210201 1        -963.40      0.0         0.0
*
*********************************************************************
*              Moderator (pipe)
*********************************************************************
*    COMPONENT NAME    COMPONENT TYPE
*
3300000 modertr           pipe
* Pipe Information  Card
*              Number Of Volumes
3300001 7
* Pipe Volume Flow Areas
*              Volume Flow AreasVol. num
3300101 13.007          7
*              Junction Flow AreasJunc. Num
3300201 13.007          6
* Pipe Volume Flow Lengths
*              Volume Flow LengthsVol. num
3300301 0.51              6
3300302 0.085             7
* Pipe Volume Flow Inclination Angles
*              Volume Flow AngleVol. Num
3300601 90.0            7
* Pipe Volume Flow Friction Data
*              Volume Roughn.Hydr. Diam.    Vol. Num
3300801 4.57e-5          1.1166             1
3300802 4.57e-5          1.1166             7
* Pipe Junction Flow friction Data
*       Forward     Reverse           Junc. Num
3300901 0.0         0.0               6
* Pipe Volume Flow Control Flags
*              pvbfe         Vol. Num
3301001 10000        7
* Pipe Junction Flow Control Flags
*              fvcahs        Junc. Num
3301101 000000       6
* Pipe Volume Flow Initial Conditions
*              ebt           pres.                          Temp
* p2nd = 15.5 bar
3301201 203      3.46697e+06    491.392      0.0      0.0      0.0      1
3301202 203      3.46231e+06    493.717      0.0      0.0      0.0      2
3301203 203      3.45765e+06    493.737      0.0      0.0      0.0      3
3301204 203      3.45300e+06    493.750      0.0      0.0      0.0      4
3301205 203      3.44834e+06    493.756      0.0      0.0      0.0      5
3301206 203      3.44369e+06    493.757      0.0      0.0      0.0      6
3301207 203      3.44097e+06    493.7561      0.0      0.0      0.0      7
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
3301300 1
* Pipe Junction Initial Conditions
*              Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
3301301    -858.04        0.0            0.0            6
```



```
*
*********************************************************************
*              Junction from moderator (330) to outlet of vessel (332).
*********************************************************************
*       COMPONENT NAME     COMPONENT TYPE
*
3310000 mod-hleg           sngljun
*        From               To                Flow area     loss coeff.    fvcahs.
3310101 332010000          330000000          0.4926         0.0  0.0       000100
*              Control var.
3310110 0.0       0.0         1.0       1.0
*              cntr.        Init.mass.flow
3310201 1        -1025.1  0.0      0.0
*
*********************************************************************
*              Outlet of the vessel.
*********************************************************************
*      COMPONENT NAME    COMPONENT TYPE
*
3320000 outlv             snglvol
*       Area  Length   Volume Azi.     Elev.     Elev(m) Roughn. Hyd.dia
3320101 0.0      0.70     0.4858  0        90.0       0.7     4.57e-6 0.47     00000
*              ebt            Pres.          Temp
3320200 203      3.46944e+06  491.393 0.0 0  0.0       0.0
*
*
*********************************************************************
*              Junction from Primary side to Outlet (negative massflow)
*********************************************************************
*         COMPONENT NAME       COMPONENT TYPE
*
3400000 out-jun               sngljun
*              From              To            Flow area     loss coeff.    cntrl var.
3400101 100000000         332000000          0.2376        0.0         0.0       0000
*              cntr.          Init.mass.flow
3400201 1        -1025.1      0.0       0.0
*
*********************************************************************
*
*      97 Fuelpins (1-19 out of 19)   (600)
*
*********************************************************************
*********************************************************************
*              BREAK, valve from upper plenum to eccs.
*********************************************************************
*       COMPONENT NAME    COMPONENT TYPE
*
6050000 porv              valve
* Valve junction geometry card
*              From              To               Junc.Area      Loss Coeff    fvcahs
6050101 317010000         610000000      1.30274e-2        0.5  0.5        000100
*              Control var.
*              Hydr. dia    beta      c           slope
6050110 0.0015      0.0        1.0       1.0
*              Control var.
6050201 0      0.0      0.0
6050300 trpvlv
6050301 502
```

```
*
**********************************************************************
*        Eccs-Pipe (610)
**********************************************************************
*     COMPONENT NAME  COMPONENT TYPE
*
6100000 Eccs          pipe
* Pipe Information  Card
*              Number Of Volumes
6100001 18
* Pipe Volume Flow Areas
*              Volume Flow AreasVol. num
*6100101 1.30274e-2              1
*6100102 2.31598e-2              2
6100102 2.31598e-2              1
6100103 7.327904e-3             2
6100104 1.287503e-2             3
6100105 8.57065e-3              4
6100106 8.633e-3                5
6100107 1.3677e-2               6
6100108 6.208e-3                9
6100109 6.12846e-3             18
*              Junction Flow AreasJunc. Num
*6100201 1.30274e-2              1
*6100202 2.31598e-2              2
6100202 2.31598e-2              1
6100203 7.327904e-3             2
6100204 1.287503e-2             3
6100205 8.57065e-3              4
6100206 8.633e-3                5
6100207 1.3677e-2               6
6100208 6.208e-3                8
6100209 6.12846e-3             17
* Pipe Volume Flow Lengths
*              Volume Flow Lengths       Vol. num
*6100301 0.047                   1
*6100302 0.025                   2
6100302 0.072                   1
6100303 0.040                   2
6100304 0.070                   3
6100305 0.05                    4
6100306 0.3                     5
6100307 1.4                     6
6100308 9.8                     7
6100309 0.3                     8
6100310 0.5                     9
6100311 1.0                    10
6100312 1.0                    18
* Pipe Volume Flow Volumes
*              Volume Flow VolumesVol. num
6100401 0.0                 18
* Pipe Volume Flow AZIMUTHAL Angles
*              Volume Flow AngleVol. Num
6100501 0.0                 18
* Pipe Volume Flow Inclination Angles
*              Volume Flow AngleVol. Num
*6100601   90.0                  18
*6100602    0.0                   2
```



```
6100602  90.0                  1
6100603   0.0                  2
6100604  90.0                  3
6100605 -60.0                  4
6100606  90.0                  5
6100607  90.0                  6
6100608   0.0                  7
6100609  90.0                  8
6100610   0.0                  9
6100611 -90.0                 11
6100612   0.0                 18
* Pipe Volume Flow Friction Data
*              Volume Roughn.Hydr. Diam.    Vol. Num
*6100801 0.1475e-6        3.0e-3          1
*6100802 0.1475e-6        4.0e-3          2
6100802 0.1475e-6        4.0e-3          1
6100803 0.1475e-6        4.5e-3          2
6100804 0.1475e-6       13.0e-3          3
6100805 0.1475e-6        7.5e-3          4
6100806 0.1475e-6        4.0e-3          5
6100807 0.1475e-6        6.0e-3          6
6100808 0.1475e-6        9.0e-3          9
6100809 0.1475e-6       51.0e-3         11
6100810 0.1475e-6       51.0e-3         18
* Pipe Volume Flow Energy Loss Coefficient
*              Forward Loss  Reverse Loss  Junc. Num
*6100901 3.0            3.0              1
*6100902 3.0            3.0              2
6100902 6.0            6.0              1
6100903 5.0            5.0              2
6100904 1.5            1.5              3
6100905 3.0            3.0              4
6100906 0.5            0.5              5
6100907 1.5            1.5              6
6100908 3.9            3.9              7
6100909 2.1            2.1              8
6100910 2.5            2.5              9
6100911 2.5            2.5             10
6100912 0.5            0.5             14
6100913 0.0            0.0             17
* Pipe Volume Flow Control Flags
*              pvbfe         Vol. Num
6101001 00000       18
* Pipe Junction Flow Control Flags
*              fvcahs        Junc. Num
6101101 000000      17
* Pipe Volume Flow Initial Conditions
*              ebt           Pres.         Temp   Vol. Num
6101201 203     3.55e+6   493.15  0.0     0.0    0.0  18
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
6101300 0
* Pipe Junction Initial Conditions
*              Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
6101301 0.0           0.0            0.0            17
*
**********************************************************************
*              BREAK, valve from upper plenum to eccs.
```

```
********************************************************************
*    COMPONENT NAME    COMPONENT TYPE
*
6980000 porv              valve
* Valve junction geometry card
*       From          To          Junc.Area Loss Coeff fvcahs subcool two-ph.
6980101 610010000   699000000   6.12846e-3  0.5  0.5 000100  0.82     1.0
*            Control var.
*            Hydr. dia   beta      c         slope
6980110 0.051      0.0     1.0      1.0
*            Control var.
6980201 0      0.0     0.0
6980300 trpvlv
6980301 502
*
********************************************************************
*            Containment (snglvol) (999).
********************************************************************
*     COMPONENT NAME    COMPONENT TYPE
*
6990000 contain          snglvol
*      Area    Length Volume Azi.Elev. Elev(m) Roughn. Hyd.dia         pvbfe
6990101 882.34   0.0  29000.0  0.0      0.0     0.0      4.57e-6 0.0      10000
*            ebt            Pres.            Temp
6990200 004      1.0e+5  293.15    0.0 0  0.0      0.0
*
********************************************************************
*            Containment (tmdpvol) (999).
********************************************************************
*          COMPONENT NAME  COMPONENT TYPE
*
*6990000 contain          tmdpvol
*      Area     Length Volume  Azi.   Elev.  Elev(m)  Roughn.Hyd.diapvbfe
*6990101 882.34     0.0  29000.0  0.0     0.0     0.0      4.57e-6 0.0       00000
*            ebt               Tripnumber
*6990200 204
*       Time Dependent Volume Data Cards
*6990201      0.0      1.0e+5  293.15  0.0
*6990202 10000.0      1.8e+5  333.15  1.0
*6990202 10000.0      1.8e+5  333.15  1.0
*
*
********************************************************************
*            Pressurizer (400).
********************************************************************
*            Pressurizer tank. (401)
********************************************************************
*          COMPONENT NAME      COMPONENT TYPE
*
4010000 przr             pipe
* Pipe Information  Card
*            Number Of Volumes
4010001 6
* Pipe Volume Flow Areas
*            Volume Flow AreasVol. num
4010101 4.9087          6
*            Junction Flow AreasJunc. Num
4010201 4.9087          5
```



```
* Pipe Volume Flow Lengths
*            Volume Flow LengthsVol. num
4010301 0.20372             2
4010302 0.61116             4
4010303 1.22232             6
* Pipe Volume Volumes
*            Volume                       Vol. num
4010401 1.0              2
4010402 3.0              4
4010403 6.0              6
* Pipe Volume Flow Inclination Angles
*            Volume Flow AngleVol. Num
4010601 90.0             6
* Pipe Volume Elevation Changes
*            Elev.                        Vol num
4010701 0.20372             2
4010702 0.61116             4
4010703 1.22232             6
* Pipe Volume Flow Friction Data
*            Volume Roughn.Hydr. Diam.   Vol. Num
4010801 4.57e-5          2.5              6
* Pipe Volume Flow Control Flags
*            pvbfe          Vol. Num
4011001 00000     6
* Pipe Junction Flow Control Flags
*            fvcahs         Junc. Num
4011101 000000  5
* Pipe Volume Flow Initial Conditions
*            ebt            pres.
4011201 200     3.40758e+06    1.00298e+06  2.41948e+06  1.76856e-07 0.0    1
4011202 200     3.40661e+06    1.00259e+06  2.43177e+06   .93769     0.0    2
4011203 200     3.40649e+06    1.00258e+06  2.43177e+06  1.0000      0.0    3
4011204 200     3.40640e+06    1.00257e+06  2.43177e+06  1.0000      0.0    4
4011205 200     3.40627e+06    1.00256e+06  2.43177e+06  1.0000      0.0    5
4011206 200     3.40609e+06    1.00255e+06  2.43177e+06  1.0000      0.0    6
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
4011300 1
* Pipe Junction Initial Conditions
*            Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
4011301 0.0              0.0              0.0           5
*
********************************************************************
*            Junction from pressurizer tank (401) to surge line (403).
********************************************************************
*       COMPONENT NAME      COMPONENT TYPE
*
4020000 przr-sl          sngljun
*         From           To              Flow area    loss coeff.    cntrl var.
4020101 403010000       401000000       0.5026548     3.0      3.0      0000
*            Control var.
4020110 0.0         0.0         1.0      1.0
*            cntr.          Init.mass.flow
4020201 1        0.0    0.0     0.0
*
********************************************************************
*            Surge Line (pipe) (403)
********************************************************************
```

```
*        COMPONENT NAME        COMPONENT TYPE
*
4030000 s-line          pipe
* Pipe Information  Card
*              Number Of Volumes
4030001 3
* Pipe Volume Flow Areas
*              Volume Flow AreasVol. num
4030101 0.15            3
*              Junction Flow AreasJunc. Num
4030201 0.15            2
* Pipe Volume Flow Lengths
*              Volume Flow LengthsVol. num
4030301 1.0             1
4030302 7.0             2
4030303 2.0             3
* Pipe Volume Flow Inclination Angles
*              Volume Flow AngleVol. Num
4030601 90.0           1
4030602  0.0           2
4030603 90.0           3
* Pipe Volume Flow Friction Data
*              Volume Roughn.Hydr. Diam.    Vol. Num
4030801 4.57e-5        0.43702          3
* Pipe Volume Flow Control Flags
*                pvbfe          Vol. Num
4031001 00000     3
* Pipe Junction Flow Control Flags
*                fvcahs         Junc. Num
4031101 000000  2
* Pipe Volume Flow Initial Conditions
*              ebt            pres.                          Temp
4031201 203        3.43181e+06      479.135   0.0    0.0      0.0      1
4031202 203        3.42715e+06      478.979   0.0    0.0      0.0      2
4031203 203        3.41781e+06      478.983   0.0    0.0      0.0      3
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
4031300 1
* Pipe Junction Initial Conditions
*              Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
4031301 0.0              0.0             0.0              2
*
*************************************************************************
*        Junction from surgeline (403) to the upper plenum (320). (404)
*************************************************************************
*        COMPONENT NAME         COMPONENT TYPE
*
4040000 prz-uppl          sngljun
*       From              To            Flow area     loss coeff.     cntrl var.
4040101 320010000        403000000        0.15            0.0     0.0       000100
*              Control var.
4040110 0.0       0.0       1.0       1.0
*              cntr.           Init.mass.flow
4040201 0          0.0       0.0      0.0
*
*************************************************************************
*              PORV, valve from pressurizer to expansion tanks.
*************************************************************************
```



```
*        COMPONENT NAME        COMPONENT TYPE
*
4050000 porv             valve
* Valve junction geometry card
*        From              To             Junc.Area      Loss Coeff        fvcahs
4050101 401010000         410000000        0.559          1000.0  1000.0      000000
*              Control var.
4050201 0      0.0      0.0
4050300 trpvlv
4050302 607
*
*************************************************************************
*              Junction from surgeline (403) to the upper plenum (320). (404)
*************************************************************************
*        COMPONENT NAME    COMPONENT TYPE
*
*4050000 prz-tmdp          sngljun
*         From              To            Flow area     loss coeff.     cntrl var.
**4050101 401010000        410000000        0.25            0.0     0.0       000100
*4050101 401010000        410000000        0.559        1000.0  1000.0      000000
*              Control var.
*4050110 0.0       0.0       1.0       1.0
*              cntr.           Init.mass.flow
*4050201 1        0.0       0.0      0.0
*
*************************************************************************
*               Expansion tanks.
*************************************************************************
*        COMPONENT NAME   COMPONENT TYPE
*
4100000 exptank          tmdpvol
* Time-dependent-volume Geometry card.
*       Area   Length Volume   Azim.    Incl.   Elev.   roughn.   hydr.    pvbfe
4100101 882.32  0.0      100.0    0.0      0.0      0.0      0.0       0.0      00010
4100200 203
4100201 0.0           1.0e+5     308.00
4100202 1000.0        1.0e+5      308.00
*
*************************************************************************
*
*              Steam Generator.  (7XX)
*
*************************************************************************
*              Separator (700).
*************************************************************************
*        COMPONENT NAME    COMPONENT TYPE
*
7000000  separatr  separatr
*  Separator Information Card
*       NJ      Initial Control Card
7000001  3        0
*  Separator Volume Geometry Cards
*      Area    Length   Volume  Azi.    Incl.    Elev(m) Roughn. Hyd.dia pvbfe
7000101 8.04     1.0      0.0     0.0      90.0      1.0    4.57e-6  1.6     00000
*  Separator Volume Initial Conditions
*      ebt       Pres.       Temp
7000200 200      1.55021e+06   8.24173e+05   2.41116e+06   1.0000
*  Vapor Outlet (N=1)
```

```
*               From              To         Area    loss coeff.    fvcahs  Void
limit
7001101 700010000  720000000  8.04     0.0  0.0      000000    0.5 *(Default)
* Liquid Fallback (N=2)
*           From     To         Area     loss coeff.    fvcahs   Void limit
7002101 700000000  707000000  1.0367   0.0  0.0     000000    0.15 *(Default)
* Inlet (N=3)
*               From              To          Area    loss coeff.    fvcahs
7003101 717010000  700000000  8.04    0.0  0.0      000000
* Vapor Outlet (N=1)
*               cntr.          Init.mass.flow
7001201          .40856       .45795            0.0
* Liquid Fallback (N=2)
*               cntr.          Init.mass.flow
7002201          .20642      -.38944          0.0
* Inlet (N=3)
*               cntr.          Init.mass.flow
7003201        .33411       .40770      0.0
*
*********************************************************************
*               Liquid Fallback (705).
*********************************************************************
*       COMPONENT NAME      COMPONENT TYPE
*
7050000 liq-fall           snglvol
*        Area     Length  Volume Azi.      Elev.    Elev(m) Roughn.Hyd.dia pvbfe
7050101 1.0367    1.0     0.0    0.0     -90.0    -1.0     4.57e-6 0.1      00000
*               ebt             Pres.           Temp
7050200 200       1.55021e+06     8.24173e+05  2.41118e+06    1.000
*
*********************************************************************
*       Junction from Downcomer to primary side tubes (711)
*********************************************************************
*       COMPONENT NAME  COMPONENT TYPE
*
7060000 Down-sec          sngljun
*         From              To               Flow area     loss coeff.      fvahs.
7060101 705010000         707000000          1.2         0.0        0.0      00000
*               cntr.          Init.mass.flow
7060201 0         .28326       5.17532e-10         0.0
*
*********************************************************************
*               Liquid Fallback (707).
*********************************************************************
*       COMPONENT NAME  COMPONENT TYPE
*
7070000 liq-fall           snglvol
*        Area     Length Volume   Azi.     Elev.  Elev(m)   Roughn. Hyd.dia pvbfe
7070101 1.0367    1.0      0.0     0.0      -90.0   -1.0     4.57e-6 0.1      00000
*               ebt             Pres.           Temp
7070200 200         1.55030e+06     8.24183e+05    2.41113e+06   0.99995
*
*********************************************************************
*               Branch between Liquid Fallback and Feedwater (708).
*********************************************************************
*               COMPONENT NAMECOMPONENT TYPE
*
7080000  fdwtr-fb   branch
```



```
* Separator Information Card
*       NJ       Initial Control Card
7080001  3        0
* Separator Volume Geometry Cards
*                Area            Length          Volume          Azi.
Incl.           Elev(m)  Roughn. Hyd.dia pvbfe
7080101 1.0367   1.0     0.0      0.0      -90.0      -1.0  4.57e-6  1.6     00000
* Separator Volume Initial Conditions
*       ebt         Pres.       Temp
7080200 200         1.55307e+06  8.20667e+05  2.41118e+06  0.42305
* Liquid Fallback In (N=1)
*               From .           To          Area     loss coeff.    fvcahs
7081101 707010002  708010001  1.0367    0.0  0.0      000101
* Water From 71506 (N=2)
*               From             To          Area     loss coeff.    fvcahs
7082101 715060006  708010003   1.0367    0.0  0.0      000103
* Liquid Out (N=3)
*               From             To          Area     loss coeff.    fvcahs
7083101 708010002  710050002   1.0367    0.0  0.0      000102
* Liquid Fallback (N=1)
*               cntr.          Init.mass.flow
7081201          .17572      -.91907   0.0
* Water From 71506 (N=2)
*               cntr.          Init.mass.flow
7082201           .50267       1.2223      0.0
* Liquid Out (N=3)
*               cntr.          Init.mass.flow
7083201          .50404       .29864       0.0
*
*********************************************************************
*               Downcomer (710)
*********************************************************************
*       COMPONENT NAME    COMPONENT TYPE
*
7100000 downcom          pipe
* Pipe Information  Card
*               Number Of Volumes
7100001 5
* Pipe Volume Flow Areas
*               Volume Flow AreasVol. num
7100101 1.2           5
*               Junction Flow AreasJunc. Num
7100201 1.2           4
* Pipe Volume Flow Lengths
*               Volume Flow LengthsVol. num
7100301 1.0             5
* Pipe Volume Flow Inclination Angles
*               Volume Flow AngleVol. Num
7100601 90.0          5
* Pipe Volume Flow Friction Data
*               Volume Roughn.Hydr. Diam.     Vol. Num
7100801 4.57e-5          0.1              5
* Pipe Volume Flow Control Flags
*               pvbfe          Vol. Num
7101001 10000         5
* Pipe Junction Flow Control Flags
*               fvcahs         Junc. Num
7101101 000000        4
```

```
* Pipe Volume Flow Initial Conditions
*              ebt              pres.                          Temp
7101201 200        1.58863e+06  8.26308e+05  2.41166e+06  9.91774e-02  0.0  1
7101202 200        1.58048e+06  8.25492e+05  2.41155e+06  .16510       0.0  2
7101203 200        1.57284e+06  8.24812e+05  2.41146e+06  .21157       0.0  3
7101204 200        1.56562e+06  8.24027e+05  2.41137e+06  .25443       0.0  4
7101205 200        1.55893e+06  8.22704e+05  2.41129e+06  .32798       0.0  5
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
*7101300 1
7101300 0
* Pipe Junction Initial Conditions
*             Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
7101301    -.30225      -.20149          0.0              1
7101302    -.31982      -.20778          0.0              2
7101303    -.33791      -.22158          0.0              3
7101304    -.37449      -.23539          0.0              4
*
*****************************************************************
*        Junction from Downcomer to Collector tubes (711)
*****************************************************************
*        COMPONENT NAME  COMPONENT TYPE
*
7110000 Down-sec          sngljun
*        From             To           Flow area      loss coeff.    fvcahs.
7110101 710000000         712000000       0.3        5.0       5.0    000000
*              cntr.           Init.mass.flow
7110201 0           1.1217         .83010   0.0
*
*
*****************************************************************
*               Water Collector, Downcomer (712)
*****************************************************************
*        COMPONENT NAME   COMPONENT TYPE
*
7120000 Collect           snglvol
*        Area  Length  Volume Azi.    Elev.   Elev(m)  Roughn.  Hyd.dia  pvbfe
7120101 1.2    1.0     0.0    0.0     -90.0   -1.0     4.57e-6  0.1      00000
*              ebt            Pres.           Temp
7120200 200      1.59480e+06 8.26418e+05    2.41173e+06     1.21769e-02
*
*****************************************************************
*        Junction from Collector to primary side tubes (713)
*****************************************************************
*       COMPONENT NAME        COMPONENT TYPE
*
7130000 Coll-sec          sngljun
*              From           To          Flow area     loss coeff.    fvcahs.
7130101 712000000         715000000       0.3        5.0       5.0    000000
*              cntr.           Init.mass.flow
7130201 0           1.0227         6.2367 0.0
*
*****************************************************************
*               Secondary side  (715)
*****************************************************************
*       COMPONENT NAME  COMPONENT TYPE
*
7150000 sec-side          pipe
```



```
* Pipe Information  Card
*                 Number Of Volumes
7150001 6
* Pipe Volume Flow Areas
*                 Volume Flow AreasVol. num
7150101 6.7876          4
7150102 8.04            6
*                 Junction Flow AreasJunc. Num
7150201 6.7876          3
7150202 8.04            5
* Pipe Volume Flow Lengths
*                 Volume Flow LengthsVol. num
7150301 1.0             5
7150302 1.0             6
* Pipe Volume Flow Inclination Angles
*                 Volume Flow AngleVol. Num
7150601 90.0          6
* Pipe Volume Flow Friction Data
*                 Volume Roughn.Hydr. Diam.    Vol. Num
*7150801 4.57e-5          0.052              4
7150801 4.57e-5          0.004              4
7150802 4.57e-5          1.6                6
* Pipe Volume Flow Control Flags
*                 pvbfe          Vol. Num
7151001 10000        6
* Pipe Junction Flow Control Flags
*                 fvcahs         Junc. Num
7151101 000000          3
7151102 001000          4
7151103 000000          5
* Pipe Volume Flow Initial Conditions
*                 ebt              pres.                          Temp
7151201 200        1.58396e+06  8.28549e+05  2.41159e+06  .23282    0.0    1
7151202 200        1.57688e+06  8.27646e+05  2.41150e+06  .31181    0.0    2
7151203 200        1.57043e+06  8.26811e+05  2.41142e+06  .38041    0.0    3
7151204 200        1.56442e+06  8.26029e+05  2.41134e+06  .41541    0.0    4
7151205 200        1.55865e+06  8.25335e+05  2.41125e+06  .39972    0.0    5
7151206 200        1.55307e+06  8.20666e+05  2.41115e+06  .42149    0.0    6
* Pipe Junction Condition Control Word
* 0=VELOCITIES, 1=MASSFLOWS
7151300 0
* Pipe Junction Initial Conditions
*                 Init.Liq.Mass Init.Vap.Mass Interface Vel.Junc. Num
7151301       5.60605e-02   .78636             0.0              1
7151302       6.03511e-02  1.0959              0.0              2
7151303       6.49180e-02  1.2717              0.0              3
7151304       5.62780e-02  1.2612              0.0              4
7151305       5.47747e-02  1.3185              0.0              5
*
*****************************************************************
*        Junction from Secondary Side to liq-steam (716)
*****************************************************************
*       COMPONENT NAME   COMPONENT TYPE
*
7160000 Down-sec          sngljun
*        From             To            Flow area    loss coeff.    fvahs.
7160101 715010000         717000000       8.04        0.0      0.0    00000
*                 cntr.          Init.mass.flow
```

```
7160201 0      -.13963      .96586      0.0
*
********************************************************************************
*              Steam and Water (717).
********************************************************************************
*      COMPONENT NAME  COMPONENT TYPE
*
7170000 stm-mix             snglvol
*       Area   Length  Volume  Azi.     Elev.    Elev(m)   Roughn.  Hyd.dia pvbfe
7170101 8.04    1.0     0.0    0.0       90.0      1.0     4.57e-6 1.6      00000
*              ebt                Pres.            Temp
7170200 200        1.55030e+06  8.24184e+05  2.41112e+06     1.0000
*
********************************************************************************
*              Steam Dome (720).
********************************************************************************
*      COMPONENT NAME       COMPONENT TYPE
*
7200000 stm-dome            snglvol
*       Area   Length  Volume  Azi.     Elev.  Elev(m)  Roughn. Hyd.dia pvbfe
7200101 9.0792   2.0     0.0    0.0     90.0    2.0     4.57e-6 1.7      00000
*         ebt            Pres.           Temp
7200200 200        1.55008e+06   8.24156e+05  2.41115e+06        1.0000
*
********************************************************************************
*              Feedwater Inlet Volume (750).
********************************************************************************
*      COMPONENT NAME      COMPONENT TYPE
*
7500000 Feedwatr            tmdpvol
*       Area    Length  Volume  Azi.  Elev.  Elev(m)  Roughn.  Hyd.dia pvbfe
7500101 4000.0   1.0     0.0    0.0    0.0     0.0     0.0       50.0     00000
*          ebt
*7500200 200
7500200 203
*       time           Pres     Temp
7500201     -1.0       15.5e+5   413.15          *t=140 C
7500202 10000.0        15.5e+5   413.15
*
********************************************************************************
*       Feedwater Inlet Junction (751)
********************************************************************************
*       COMPONENT NAME  COMPONENT TYPE
*
7510000 Feedwtr             tmdpjun
*              From       To                Flow area
7510101 750000000         715060004          0.02
*              cntr.            Init.mass.flow
7510200 1    503
*        time         liq.mass   stm.mass   inter.mass
7510201     -1.0      32.140       0.0        0.0
7510202      0.0      0.0         0.0        0.0
7510203 10000.0       0.0         0.0        0.0
*
********************************************************************************
*              Outlet from steam generator to steam line.
********************************************************************************
*        COMPONENT NAME  COMPONENT TYPE
```



```
*
7600000 s-gvalve              valve
* Valve junction geometry card
*        From              To               Junc.Area      Loss Coeff       fvcahs
7600101 720010000          761000000         1.12567          0.0  0.0        000000
*              Control var.
7600201 0      3.271     3.271   0.0
7600300 trpvlv
7600302 603
*
********************************************************************************
*              Steam Line (761).
********************************************************************************
*        COMPONENT NAME    COMPONENT TYPE
*
7610000 Stmline             tmdpvol
*        Area    Length  Volume Azi.     Elev.    Elev(m)Roughn.   Hyd.dia pvbfe
7610101 4000.0   1.0      0.0    0.0      0.0      0.0     0.0      50.0     00000
*        ebt
7610200 200
*        time           Pres     Temp
7610201     0.0         15.5e+5     825.91e+3  2411.20e3       1.000
7610202   300.0         15.5e+5     825.91e+3  2411.20e3       1.000
7610203  1000.0         15.5e+5     825.91e+3  2411.20e3       1.000
*
********************************************************************************
*              SRV, valve from steam generator to expansion tanks.
********************************************************************************
*        COMPONENT NAME      COMPONENT TYPE
*
7700000 s-gvalve              valve
* Valve junction geometry card
*       From              ToJunc.               Area        Loss Coeff       fvcahs
7700101 720010000         780000000            0.559         1.0  1.0        000000
*              Control var.
7700201 0       0.0       0.0
7700300 trpvlv
7700302 612
*
********************************************************************************
*              Containment (780).
********************************************************************************
*        COMPONENT NAME     COMPONENT TYPE
*
7800000 Contain3            tmdpvol
*        Area    Length  Volume Azi.     Elev.    Elev(m)Roughn.   Hyd.dia pvbfe
7800101 40.0     1.0      0.0    0.0      0.0      0.0     0.0      50.0     00000
*.             ebt
7800200 203
*        time           Pres     Temp
7800201     0.0         1.0e+5   308.00
7800202 10000.0         1.0e+5   308.00
*
********************************************************************************
*
*      Heat structure input (11051 - 13302)
*
********************************************************************************
```

```
*****************************************************************
*      Hot Leg Pipe (11001)
*****************************************************************
*
* General Heat Structure Input
*        NH  NP  Geom.type  Init.flag  left.coor Reflood Bound.Vol Max.number.
11001000   5   2   2          1          0.1375       0       1           16
* Heat Structure Mesh Flags
*         mesh loc. flag
11001100    0       2
* Heat Structure Mesh Interval Data, Radial Information
*        Mesh Interval     Interval
11001101  0.012          1
* Heat Structure Composition Data, Radial Information
*         MMM              Interval
11001201  300              1
* Heat Structure Source Distribution Date.
*         Source Value     Interval
11001301  0.0              1
* Initial Temperature Flag
*         0/-1
11001400  -1
* Initial Temperature Data
*         radius1   radius2
11001401  491.38   491.38
11001402  491.70   491.70
11001403  491.72   491.72
11001404  491.73   491.73
11001405  491.73   491.73
* Left Boundary Condition Card
*        Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
11001501  100010000      0      1                  1               4.2     1
11001502  100020000      0      1                  1               1.4142  2
11001503  100030000      0      1                  1               3.2     3
11001504  100040000      0      1                  1               1.4142  4
11001505  100050000      0      1                  1               2.2     5
* Right Boundary Condition Card
*        Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
11001601  0              0          0                 1            4.2     1
11001602  0              0          0                 1            1.4142  2
11001603  0              0          0                 1            3.2     3
11001604  0              0          0                 1            1.4142  4
11001605  0              0          0                 1            2.2     5
*
* Source Data Cards
*        Sourcetype  Mult  Direct Multi. Heat (L/R)         #
11001701  0      0.0      0.0      0.0             5
*
* Aditional Left Boundary Cards
*       Eqdia     len.for. len rev.  grid len for rev   loss coeff Loc boil  #
11001801 0.275   2.1      2.1      0.0      .0       .0   .0    0            1
11001802 0.275   0.705    0.705    0.0      .0       .0   .0    0            2
11001803 0.275   1.6      1.6      0.0      .0       .0   .0    0            3
11001804 0.275   0.705    0.705    0.0      .0       .0   .0    0            4
11001805 0.275   1.1      1.1      0.0      .0       .0   .0    0            5
*
*****************************************************************
*      Steam Generator Pipes (11051)
```



```
*****************************************************************
*
* General Heat Structure Input
*         NH  NP  Geom.type  Init.flag  left.coor Reflood Bound.Vol Max.number.
11051000  8   2   2          1          0.0041       0       1           16
* Heat Structure Mesh Flags
*          mesh loc. flag
11051100 . 0       2
* Heat Structure Mesh Interval Data, Radial Information
*         Mesh Interval     Interval
11051101  0.0010            1
* Heat Structure Composition Data, Radial Information
*         MMM               Interval
11051201  300              1
* Heat Structure Source Distribution Date.
*          Source Value     Interval
11051301  0.0               1
* Initial Temperature Flag
**          0/-1
11051400  -1
* Initial Temperature Data
*          radius1   radius2
11051401  483.60    478.76
11051402  481.56    477.88
11051403  479.96    477.13
11051404  478.69    476.48
11051405  477.75    476.07
11051406  477.12    475.90
11051407  476.67    475.79
11051408  476.38    475.76
* Left Boundary Condition Card
*         Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
11051501   105010000     10000      1               1             7976.0  8
* Right Boundary Condition Card
*         Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
11051601  715010000    10000       1                1             7976.0  4
11051602  715040000    -10000      1                1             7976.0  8
* Source Data Cards
*         Sourcetype  Mult  Direct Multi. Heat (L/R)        #
11051701  0     0.0     0.0       0.0             8
*
* Additional Left Boundary Cards
*         Eqdia     len.for. len rev.  grid len for rev   loss coeff Loc boil  #
11051801 0.0082   0.5      0.5      0.0      .0        .0   .0    0        8
*
* Additional Right Boundary Cards
*         Eqdia     len.for. len rev.  grid len for rev   loss coeff Loc boil  #
11051901 0.004    0.5      0.5      0.0      .0        .0   .0    0        8
*
*****************************************************************
*      Suction Leg Pipe (11101)
*****************************************************************
*
* General Heat Structure Input
*         NH  NP  Geom.type  Init.flag  left.coor Reflood Bound.Vol Max.number.
11101000   5   2   1          1          0.1375       0       1           16
* Heat Structure Mesh Flags
*          mesh loc. flag
```

```
11101100    0       2
* Heat Structure Mesh Interval Data, Radial Information
*         Mesh Interval    Interval
11101101  0.012            1
* Heat Structure Composition Data, Radial Information
*          MMM             Interval
11101201  300              1
* Heat Structure Source Distribution Date.
*         Source Value     Interval
11101301  0.0              1
* Initial Temperature Flag
*         0/-1
11101400  -1
* Initital Temperature Data
*         radius1   radius2
11101401  481.38    481.38
11101402  481.70    481.70
11101403  481.72    481.72
11101404  481.73    481.73
11101405  481.73    481.73
* Left Boundary Condition Card
*        Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
11101501  110010000       0        1                1                  2.2   1
11101502  110020000       0        1                1                  1.4142  2
11101503  110030000       0        1                1                  1.3   3
11101504  110040000       0        1                1                  1.4142  4
11101505  110050000       0        1                1                  1.0   5
* Right Boundary Condition Card
*        Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
11101601  0               0        0                1                  2.2   1
11101602  0               0        0                1                  1.4142  2
11101603  0               0        0                1                  1.3   3
11101604  0               0        0                1                  1.4142  4
11101605  0               0        0                1                  1.0   5
*
* Source Data Cards
*         Sourcetype  Mult  Direct Multi. Heat (L/R)          #
11101701  0     0.0       0.0          0.0                 5
*
* Aditional Left Boundary Cards
*        Eqdia    len.for. len rev.  grid len for rev   loss coeff Loc boil  #
11101801 0.275    1.1      2.1       0.0      .0         .0   .0    0       1
11101802 0.275    0.705    0.705     0.0      .0         .0   .0    0       2
11101803 0.275    0.65     0.65      0.0      .0         .0   .0    0       3
11101804 0.275    0.705    0.705     0.0      .0         .0   .0    0       4
11101805 0.275    0.5      0.5       0.0      .0         .0   .0    0       5
*
*******************************************************************
*       Fuel Rods (13171)
*******************************************************************
*
* General Heat Structure Input
*        NH  NP  Geom.type  Init.flag  left.coor Reflood Bound.Vol Max.number.
13171000  6   7   2          1          0.0       0       1         16
* Gap Conductance Mode Initial Gap pressure Data
*          P      Ref.Volume
13171001  2.0e5   317070000
* Metal-Water Reaction Control Card
```



```
*          OXTO
13171003  1.0e-9
* Gap Deformation Data
*          Fuel roughn.  Cladding roughn.  displacement
13171011  1.0e-6        2.0e-6            1.0e-6 -1.0e-6       6
* Heat Structure Mesh Flags
*          mesh loc. flag
13171100  0      2
* Heat Structure Mesh Interval Data, Radial Information
*           Mesh Interval    Interval
13171101  0.00215           4
13171102  0.0002            5
13171103  0.0009            6
* Heat Structure Composition Data, Radial Information
*           MMM             Interval
13171201  100               4
13171202  150               5
13171203  200               6
* Heat Structure Source Distribution Date.
*           Source Value    Interval
13171301  1.0               4
13171302  0.0               5
13171303  0.0               6
* Initial Temperature Flag
*           0/-1
13171400  -1
* Initital Temperature Data
*          Temp  Interval
13171401    997.84    975.01    908.11   826.67   722.97   506.89   490.86
13171402   1001.72    978.83    911.75   830.07   725.97   510.75   494.76
13171403   1004.81    981.88    914.65   832.76   728.36   513.80   497.86
13171404   1007.87    983.89    917.52   835.43   730.72   516.81   500.91
13171405   1010.87    987.85    920.33   838.05   733.08   519.82   503.96
13171406    958.47    950.43    926.74   788.56   725.50   509.22   500.36
* Left Boundary Condition Card
*        Bound.vol.no Increment Bound.cond.type surface.area.code  Area  #
13171501  0            0         0               1                  921.5  6
* Right Boundary Condition Card
*        Bound.vol.no Increment Bound.cond.type surface.area.code  Area  #
13171601  317010000    10000     1               1                  921.5  6
* Source Data Cards
*         Sourcetype   Mult  Direct Multi. Heat (L/R)        #
13171701  999          0.175   0.0          0.0              5
13171702  999          0.125   0.0          0.0              6
* Aditional Right Boundary Cards
*         Eqdia    len.for. len rev.  grid len for rev   loss coeff Loc boil  #
13171901 0.0186   0.255     0.255     0.0      .0         .0   .0    0      6
*
*******************************************************************
*       Vessel Internals (13301)
*******************************************************************
*
* General Heat Structure Input
*         NH  NP  Geom.type  Init.flag  left.coor Reflood Bound.Vol Max.number.
13301000   7   3   1          1          4.2       0       1         16
* Heat Structure Mesh Flags
*          mesh loc. flag
13301100   0      2
```

```
* Heat Structure Mesh Interval Data, Radial Information
*        Mesh Interval     Interval
13301101  0.075            1
13301102  0.100            2
* Heat Structure Composition Data, Radial Information
*         MMM              Interval
13301201  300              2
* Heat Structure Source Distribution Date.
*         Source Value     Interval
13301301  0.0              2
* Initial Temperature Flag
*         0/-1
13301400  -1
* Initital Temperature Data
*         radius1   radius2
13301401   491.38   491.38  491.38
13301402   493.70   493.70  493.70
13301403   493.72   493.72  493.72
13301404   493.73   493.73  493.73
13301405   493.74   493.74  493.74
13301406   493.74   493.74  493.74
13301407   493.74   493.74  493.74
* Left Boundary Condition Card
*        Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
13301501  330010000     10000      1                 0                7.5   6
13301502  330070000     0          1                 0                3.275 7
* Right Boundary Condition Card
*        Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
13301601  0             0          0                 0                7.5   6
13301602  0             0          0                 0                3.275 7
* Source Data Cards
*        Sourcetype  Mult  Direct Multi. Heat (L/R)        #
13301701   0          0.0     0.0          0.0              7
*
* Aditional Left Boundary Cards
*       Eqdia     len.for. len rev.  grid len for rev   loss coeff Loc boil  #
13301801 1.1166   0.255      0.255      0.0       .0        .0   .0   0      6
13301802 1.1166   0.0425     0.0425     0.0       .0        .0   .0   0      7
*
*************************************************************************
*     Vessel Internals (13302), Fuel element tubes.
*************************************************************************
*
* General Heat Structure Input
*       NH  NP  Geom.type  Init.flag  left.coor Reflood Bound.Vol Max.number.
13302000  7   3   2          1          0.0       0       1         16
* Heat Structure Mesh Flags
*         mesh loc. flag
13302100  0        2
* Heat Structure Mesh Interval Data, Radial Information
*        Mesh Interval     Interval
13302101  0.000375         2
* Heat Structure Composition Data, Radial Information
*         MMM              Interval
13302201  200              2
* Heat Structure Source Distribution Date.
*         Source Value     Interval
13302301  0.0              2
```



```
* Initial Temperature Flag
*         0/-1
13302400  -1
* Initital Temperature Data
*         radius1    radius2
13302401  480.47       481.00       481.53
13302402  483.55       484.03       484.51
13302403  486.38       486.69       487.00
13302404  489.21       489.37       489.53
13302405  492.09       492.13       492.17
13302406  493.74       493.74       493.74
13302407  493.74       493.74       493.74
* Left Boundary Condition Card
*        Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
13302501  317010000     10000      1                 1               5.063  6
13302502  317070000     0          1                 1               2.532  7
* Right Boundary Condition Card
*        Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
13302601  330010000     10000      1                 1               5.063  6
13302602  330070000     0          1                 1               2.532  7
* Source Data Cards
*        Sourcetype  Mult  Direct Multi. Heat (L/R)        #
13302701   0          0.0     0.0          0.0              7
*
* Aditional Left Boundary Cards
*       Eqdia    len.for. len rev.  grid len for rev   loss coeff Loc boil  #
13302801 0.0186   0.5       0.5      0.0       .0        .0   .0   0       6
13302802 0.0186   0.0425    0.5      0.0       .0        .0   .0   0       7
*
* Aditional Right Boundary Cards
*       Eqdia    len.for. len rev.  grid len for rev   loss coeff Loc boil  #
13302901 1.1166   0.5       0.5      0.0       .0        .0   .0   0       6
13302902 1.1166   0.0425    0.5      0.0       .0        .0   .0   0       7
*
*************************************************************************
*     Pressurizer walls (14011)
*************************************************************************
*
* General Heat Structure Input
*       NH  NP  Geom.type  Init.flag  left.coor Reflood Bound.Vol Max.number.
14011000  6   2   2          1          1.25      0       1         16
* Heat Structure Mesh Flags
*         mesh loc. flag
14011100    0      2
* Heat Structure Mesh Interval Data, Radial Information
*        Mesh Interval     Interval
14011101  0.050            1
* Heat Structure Composition Data, Radial Information
*         MMM              Interval
14011201  300              1
* Heat Structure Source Distribution Date.
*         Source Value     Interval
14011301  0.0              1
* Initial Temperature Flag
*         0/-1
14011400  -1
* Initital Temperature Data
*         radius1    radius2
```

```
14011401     513.45   513.45
14011402     513.45   513.45
14011403     513.45   513.45
14011404     513.45   513.45
14011405     513.45   513.45
14011406     513.45   513.45
* Left Boundary Condition Card
*         Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
14011501  401010000       10000      1                 1         0.20372    2
14011502  401030000       10000      1                 1         0.61116    4
14011503  401050000       10000      1                 1         1.22232    6
* Right Boundary Condition Card
*         Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
14011601  0               0          0               1         0.20372      2
14011602  0               0          0               1         0.61116      4
14011603  0               0          0               1         1.22232      6
*
* Source Data Cards
*         Sourcetype  Mult  Direct Multi. Heat (L/R)      #
14011701  0     0.0     0.0          0.0             6
*
* Aditional Left Boundary Cards
*       Eqdia    len.for. len rev.  grid len for rev    loss coeff Loc boil  #
14011801 2.5    0.1       0.1    0.0       .0          .0   .0    0         2
14011802 2.5    0.305     0.305  0.0       .0          .0   .0    0         4
14011803 2.5    0.61      0.61   0.0       .0          .0   .0    0         6
*
*****************************************************************
*     Surge Line Pipe (14031)
*****************************************************************
*
* General Heat Structure Input
*         NH  NP  Geom.type  Init.flag  left.coor Reflood Bound.Vol Max.number.
14031000   3   2   1           1          0.2175       0        1        16
* Heat Structure Mesh Flags
*          mesh loc. flag
14031100     0      2
* Heat Structure Mesh Interval Data, Radial Information
*         Mesh Interval    Interval
14031101  0.012              1
* Heat Structure Composition Data, Radial Information
*         MMM              Interval
14031201  300                1
* Heat Structure Source Distribution Date.
*          Source Value    Interval
14031301  0.0                1
* Initial Temperature Flag
*          0/-1
14031400  -1
* Initital Temperature Data
*          radius1   radius2
14031401  478.38    478.38
14031402  478.70    478.70
14031403  478.72    478.72
* Left Boundary Condition Card
*         Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
14031501  403010000       0      1                   1           1.0      1
14031502  403020000       0      1                   1           7.0      2
```



```
14031503  403030000       0      1                   1           2.0      3
* Right Boundary Condition Card
*          Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
14031601  0               0          0                1            1.0     1
14031602  0               0          0                1            7.0     2
14031603  0               0          0                1            2.0     3
*
* Source Data Cards
*          Sourcetype  Mult  Direct Multi. Heat (L/R)       #
14031701  0     0.0      0.0        0.0            3
*
* Aditional Left Boundary Cards
*        Eqdia    len.for. len rev.  grid len for rev    loss coeff Loc boil  #
14031801 0.43702   0.5       0.5     0.0       .0          .0   .0   0        1
14031802 0.43702   3.5       3.5     0.0       .0          .0   .0   0        2
14031803 0.43702   0.5       0.5     0.0       .0          .0   .0   0        3
*
*****************************************************************
*     S.G. Internals (17101).
*****************************************************************
*
* General Heat Structure Input
*        NH  NP  Geom.type  Init.flag  left.coor Reflood Bound.Vol Max.number.
17101000  5   3   2            1          0.8       0        1        16
* Heat Structure Mesh Flags
*           mesh loc. flag
17101100  0      2
* Heat Structure Mesh Interval Data, Radial Information
*          Mesh Interval    Interval
17101101  0.005           2
* Heat Structure Composition Data, Radial Information
*          MMM              Interval
17101201  300                2
* Heat Structure Source Distribution Date.
*           Source Value    Interval
17101301  0.0                2
* Initial Temperature Flag
*           0/-1
17101400  -1
* Initital Temperature Data
*           radius1   radius2
17101401  473.38      473.32      473.27
17101402  473.72      473.68      473.64
17101403  473.86      473.82      473.78
17101404  474.00      473.96      473.92
17101405  474.18      474.14      474.09
* Left Boundary Condition Card
*          Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
17101501  715050000     -10000       1                1            1.0     5
* Right Boundary Condition Card
*          Bound.vol.no  Increment  Bound.cond.type  surface.area.code  Area  #
17101601  710050000     -10000       1                1            1.0     5
* Source Data Cards
*          Sourcetype  Mult  Direct Multi. Heat (L/R)       #
17101701   0          0.0      0.0       0.0             5
*
* Aditional Left Boundary Cards
*         Eqdia    len.for. len rev.  grid len for rev   loss coeff Loc boil  #
```

```
17101801 0.0040   0.5     0.5     0.0     .0      .0  .0  0     4
17101802 1.6000   0.5     0.5     0.0     .0      .0  .0  0     5
*
* Aditional Right Boundary Cards
*       Eqdia    len.for. len rev.  grid len for rev   loss coeff Loc boil #
17101901 0.1000   0.5     0.5     0.0     .0      .0  .0  0     5
*
**********************************************************************
*    Heat Structure Thermal Property Data
**********************************************************************
*
*  Internal Data Tables
20110000     tbl/fctn   1     1      *100, UO2
20120000     tbl/fctn   1     1      *200, Zr
20130000     tbl/fctn   1     1      *300, s-steel
*
*  Gap Conductance Model      *150
20115000     tbl/fctn  3  1
20115001     helium     0.90
20115002     nitrogen   0.10
20115051     2.6e3
*
**********************************************************************
*    Table no 100:  Thermal Conductivity, UO2
*
20110001     2.731500e+02  5.782385
20110002     5.331500e+02  5.782385
20110003     6.164833e+02  4.622792
20110004     6.998167e+02  4.633177
20110005     7.831500e+02  4.221262
20110006     8.664833e+02  3.880307
20110007     9.498167e+02  3.596467
20110008     1.033150e+03  3.357625
20110009     1.088706e+03  3.155129
20110010     1.199817e+03  2.983787
20110011     1.283150e+03  2.836674
20110012     1.366483e+03  2.713792
20110013     1.449817e+03  2.608217
20110014     1.533150e+03  2.521680
20110015     1.616483e+03  2.448990
20110016     1.699817e+03  2.391875
20110017     1.977594e+03  2.289762
20110018     2.255372e+03  2.307069
20110019     2.533150e+03  2.433413
20110020     2.810928e+03  2.661870
20110021     3.088706e+03  2.994171
*
**********************************************************************
*    Table no 100:  volumetric heat capacity, UO2
*
20110051     2.731500e+02  2.310427e+06
20110052     3.231500e+02  2.571985e+06
20110053     3.731500e+02  2.746357e+06
20110054     4.731500e+02  2.920729e+06
20110055     6.731500e+02  3.138694e+06
20110056     1.373150e+03  3.443844e+06
20110057     1.773150e+03  3.531030e+06
20110058     1.973150e+03  3.792588e+06
```



```
20110059     2.173150e+03  4.228518e+06
20110060     2.373150e+03  4.882412e+06
20110061     2.673150e+03  6.015829e+06
20110062     2.773150e+03  6.320980e+06
20110063     2.873150e+03  6.582538e+06
20110064     2.973150e+03  6.713317e+06
20110065     3.113150e+03  6.800503e+06
20110066     4.699817e+03  6.800503e+06
*
**********************************************************************
*    Table no 200:  Thermal Conductivity, Zr
*
20120001     2.731500e+02  1.000438e+01
20120002     4.731500e+02  1.200438e+01
20120003     6.731500e+02  1.400510e+01
20120004     8.731500e+02  1.700793e+01
20120005     1.073150e+03  1.900866e+01
20120006     1.273150e+03  2.200975e+01
20120007     1.473150e+03  2.501085e+01
20120008     1.673150e+03  3.001267e+01
20120009     1.873150e+03  3.601486e+01
20120010     2.073150e+03  4.401777e+01
20120011     2.273150e+03  5.502352e+01
20120012     2.473150e+03  6.802826e+01
*
**********************************************************************
*    Table no 200:  volumetric heat capacity, Zr
*
20120051    2.553722e+02   1.904141e+06
20120052    1.077594e+03   2.312171e+06
20120053    1.185928e+03   5.712422e+06
20120054    1.248428e+03   2.311769e+06
20120055    2.199817e+03   2.312171e+06
*
**********************************************************************
*    Table no 300:  Thermal Conductivity, S-steel
*
20130001     2.731500e+02    1.298051e+01
20130002     1.199817e+03    2.510604e+01
*
**********************************************************************
*    Table no 300:  volumetric heat capacity, S-steel
*
20130051      2.664833e+02    3.830413e+06
20130052      3.664833e+02    3.830413e+06
20130053      4.220389e+02    3.964814e+06
20130054      4.775944e+02    4.099214e+06
20130055      5.331500e+02    4.233615e+06
20130056      5.887056e+02    4.334415e+06
20130057      6.442611e+02    4.435081e+06
20130058      6.998167e+02    4.502416e+06
20130059      8.109278e+02    4.636816e+06
20130060      1.366483e+03    5.376019e+06
*
**********************************************************************
*   General Tables  Scram curve
**********************************************************************
*    Table no 999:  Power
```

```
20299900      power      503
20299901       -1.0         65.0e+06
20299902        0.0         65.0e+06
20299903        0.3         61.35e+06
20299904        0.6         58.73e+06
20299905        1.0         54.59e+06
20299906        3.0         34.00e+06
20299907        6.0          3.34e+06
20299908       10.0          3.11e+06
20299909       30.0          2.63e+06
20299910       60.0          2.33e+06
20299911      100.0          2.11e+06
20299912      300.0          1.72e+06
20299913      600.0          1.50e+06
20299914     1000.0          1.32e+06
20299915     3000.0          0.954e+06
20299916     6000.0          0.764e+06
20299917    10000.0          0.659e+06
20299918    30000.0          0.524e+06
20299919    60000.0          0.436e+06
20299920   100000.0          0.379e+06
20299921   300000.0          0.259e+06
20299922   600000.0          0.193e+06
.end of input
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STATENS KÄRNKRAFTINSPEKTION
Swedish Nuclear Power Inspectorate

POST/POSTAL ADDRESS SE-106 58 Stockholm
BESÖK/OFFICE Klarabergsviadukten 90
TELEFON/TELEPHONE +46 (0)8 698 84 00
TELEFAX +46 (0)8 661 90 86
E-POST/E-MAIL ski@ski.se
WEBBPLATS/WEB SITE www.ski.se