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ABSTRACT

This project was performed within the fifth and final phase of sub-project RAK-2.1 of the Nordic Co-operative Reactor Safety Program, NKS. RAK-2.1 has also included studies of reflooding of degraded core, recriticality and late phase melt progression. Earlier source term calculations for Swedish nuclear power plants are based on the integral code MAAP. A need was recognised to compare these calculations with calculations done with mechanistic codes. In the present work SCDAP/RELAP5 and CONTAIN were used.

Only limited results could be obtained within the frame of RAK-2.1, since many problems were encountered using the SCDAP/RELAP5 code. The main obstacle was the extremely long execution times of the MOD3.1 version, but also some dubious fission product calculations. However, some interesting results were obtained for the studied sequence, a total loss of AC power. The report describes the modelling approach for SCDAP/RELAP5 and CONTAIN, and discusses results for the transient including the event of a surge line creep rupture.

The study will probably be completed later, providing that an improved SCDAP/RELAP5 code version becomes available.

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EXECUTIVE SUMMARY

The aim of this study, done within the Nordic RAK2.1 project, was to calculate the source term for the Swedish nuclear power plant Ringhals 2, a Westinghouse, 3-loop PWR. The source term describes the magnitude, duration and physical and chemical forms of a release of radioactive fission products from the containment to the environment. These kinds of calculations have so far only been done, in Sweden, with the integral code MAAP, and a need was recognised to compare these calculations with calculations done with mechanistic codes.

The initial core inventory of fission products was calculated with ORIGEN2. The melting of the core, the vessel failure and the transport of fission products within the primary system was calculated with SCDAP/RELAP5/Mod3.1E and COUPLE, and for the thermalhydraulics and the fission products in the containment CONTAIN was used. To use SCDAP/RELAP5 for fission product transport calculations was a new application for the code, there was none or little experience to rely on. The code was not well documented as far as the fission product transport was concerned, but was said to handle this.

Initially, several sequences were planned, but as the problems with the code increased, there was only time to study one sequence. The initial event was a total black out, and in addition a mal-functioning auxiliary feedwater pump. When the steam generators were dried out, there was no cooling possibility left, and the core would melt and eventually the vessel would be melt through. Releases from the primary system will initially occur through valves connected to the pressurizer and possibly from breaks in the surge line, and eventually a big release when the vessel is melt through. The mass of fission products released from the primary system would be recorded and used as input to CONTAIN, and the distribution of fission products in the containment and the releases to environment could be calculated.

A SCDAP/RELAP5 model of Ringhals 2 was made. When calculations started it became obvious that the fission product transport routine did not work properly and the calculations had to be interrupted. No final results could be reached within the time frame of the RAK-2.1 project, to a great extent also due to the extremely long execution times for the code. The project will now pause until another release of SCDAP/RELAP5 is obtained in which these problems are solved.

SAMMANFATTNING

Målet med detta arbetet, som varit ett delprojekt i NKS/RAK2.1, var att beräkna källtermen för det svenska kärnkraftverket Ringhals 2, PWR. Källtermen beskriver storleken och varaktigheten såväl som den fysikaliska och kemiska sammansättningen hos ett utsläpp av radioaktiva fissionsprodukter. Avsikten var att studera källtermen i inneslutning och till omgivningen. Den här sortens beräkningar har hittills bara gjorts med det integrerade programmet MAAP i Sverige, och en alternativ, jämförande studie med ett mer detaljerat och mekanistiskt system av program var önskvärd.

Arbetet började med en beräkning av härdinventariet med hjälp av ORIGEN2. Härdsmältan, tankbrott och transport av fissionsprodukter i primärsystemet skulle beräknas med SCDAP/RELAP5/Mod3.1E och COUPLE, och det termohydrauliska förloppet och fissionsprodukternas spridning i inneslutningen skulle beräknas med CONTAIN. När arbetet startade fanns mycket liten erfarenhet av denna slags beräkningar med SCDAP/RELAP5.

Ursprungligen var ett flertal sekvenser planerade, men allteftersom problemen med SCDAP/RELAP5 hopade sig koncentrerades arbetet till en sekvens. Den innebar initialt totalt elbortfall och en felande hjälpmavapump. När ånggeneratorerna tömts, finns ingen kylningsmöjlighet kvar, vilket så småningom leder till att härden börjar smälta. Nedfallen härd kommer sedan att smälta igenom tankbotten. Utsläpp till inneslutningen kommer till att börja med att ske genom ventiler på tryckhållaren och eventuellt genom krypbrott i tryckhållarledningen. Det största utsläppet uppstår när härden smälter igenom tankbotten. Mängden utsläpp till inneslutningen skulle beräknas med SCDAP/RELAP5 och användas som indata till CONTAIN, som sen skulle beräkna fördelningen i inneslutningen och hur mycket som släpps ut till omgivningen.

Initialt lades mycket tid ner på att göra en för dessa beräkningar relevant indatafil för Ringhals 2. När beräkningarna med SCDAP/RELAP5 började blev det snart uppenbart att programmet inte behandlade beräkningar av fissionsprodukter korrekt, och arbetet avbröts. Förseningen i arbetet vållades också genom de extremt långa beräkningstiderna, i storleksordningen veckor, för koden. Inga slutliga resultat kunde nås. Projektet gör nu ett uppehåll i avvaktan på en ny kodversion där, förhoppningsvis, dessa problem lösts.

1 INTRODUCTION

The present work aimed at studying the source term to containment and to environment at total loss of power. The thermalhydraulic conditions and severe accident progression in the primary system and in the containment were studied for the Swedish PWR plant Ringhals 2. The calculations were done according to "best estimate" and not in a conservative way.

The initial core inventory was calculated with ORIGEN2. The primary and secondary system was modelled in SCDAP/RELAP5, and this output was then taken as input to CONTAIN and the containment model.

In Chapter 2 the term "source term" is described and references to old source term calculations are given. The Ringhals 2-model is described in Chapter 3, and the sequence studied is discussed in Chapter 4. Results are given in Chapter 5 and conclusions in Chapter 6.

2 BACKGROUND

Many radioactive materials are produced from nuclear fission in the reactor core, e.g. fission products which can be in states of non-volatile, volatile and noble gases. During normal conditions these are collected inside the fuel or in the gap in-between the fuel pellets and the cladding. If the fuel rod cladding is ruptured, these are able to escape into the coolant. If leaks from the primary system exist they can escape into the containment and further to the environment. The term "source term" has come to mean the magnitude and duration of such a release, and has also come to include a description of the physical and chemical forms of the released materials.

A first attempt to estimate the source term was made in TID-14844 [1]. It was assumed that 100 % of the core noble gases, 50 % of the core halogens and 1 % of the solid fission products were released to the containment. It was pointed out in the report though that these were approximations only.

In 1974 WASH-1400 [2] was published, which was the first systematic attempt to estimate the source term. The aim of the report was to make an assessment of the risks of commercial nuclear power plants to the public, and it is shown that the greatest risks are associated with beyond-design-basis accidents. Releases to fuel rod gas gap and to coolant are estimated, as well as releases to containment and to surrounding soil. But some important phenomena were neglected and the assumptions made were not always realistic.

The TMI-2 accident became a turning point. The releases were unexpectedly low. The methods which the source term estimates so far had been based upon had to be reassessed. It was discovered that certain important phenomena had been neglected, such as the fission product vapour and aerosol chemistry and the vapour and aerosol retention. Reference [3] is a state-of-the art report from 1994, describing in detail what has happened since the TMI-2 accident, how this field of research has been evolving.

3 MODEL

The Ringhals 2 power plant is a three loop, two turbine PWR of Westinghouse Stal-Laval design with ASEA electrical generators. During the summer 1989 the original steam generators were replaced with new ones of KWU design. The nominal thermal power is 2 660 MW.

The used SCDAP/RELAP5 model was based on an already existing Ringhals 2 RELAP5 model, that had been used in previous analyses. A SCDAP input was added and the core and the power operated relief valves were modelled in greater detail.

3.1 Nodalization of primary and secondary system

The nodalization of the core is based on the results from the PWR nodalization sensitivity studies made in the SCDAP/RELAP5 manual, volume V [4]. Thus, the core is divided into ten axial nodes and five radial channels, see Figures 1 and 3. Ten axial nodes give almost the same results as a 20-node-division and is said to have notably shorter run-times. Five radial channels are considered better than three radial channels, when a surge line rupture is modelled and the timing of the core heat-up is important.

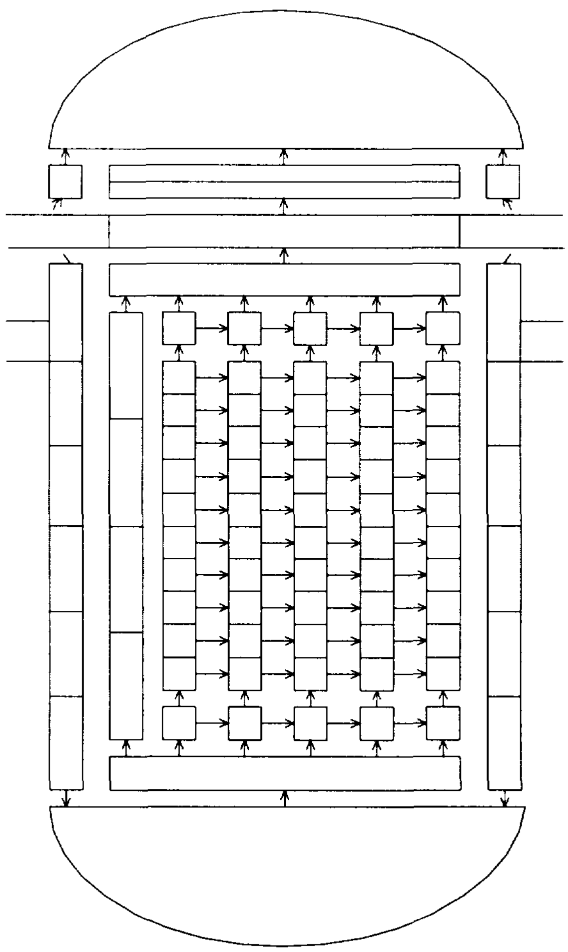

Figure 1. Nodalization of the vessel.

The rest of the primary and secondary systems is modelled according to Figure2. Ringhals 2 has got three steam generators. Two of them are added into one double loop and the third steam generator makes up one single loop on its own.

All heat structures are modelled, too, although they are not always shown in the figures. Structures in the core are modelled in SCDAP as core components, the others in primary and secondary systems, as RELAP5 heat structures.

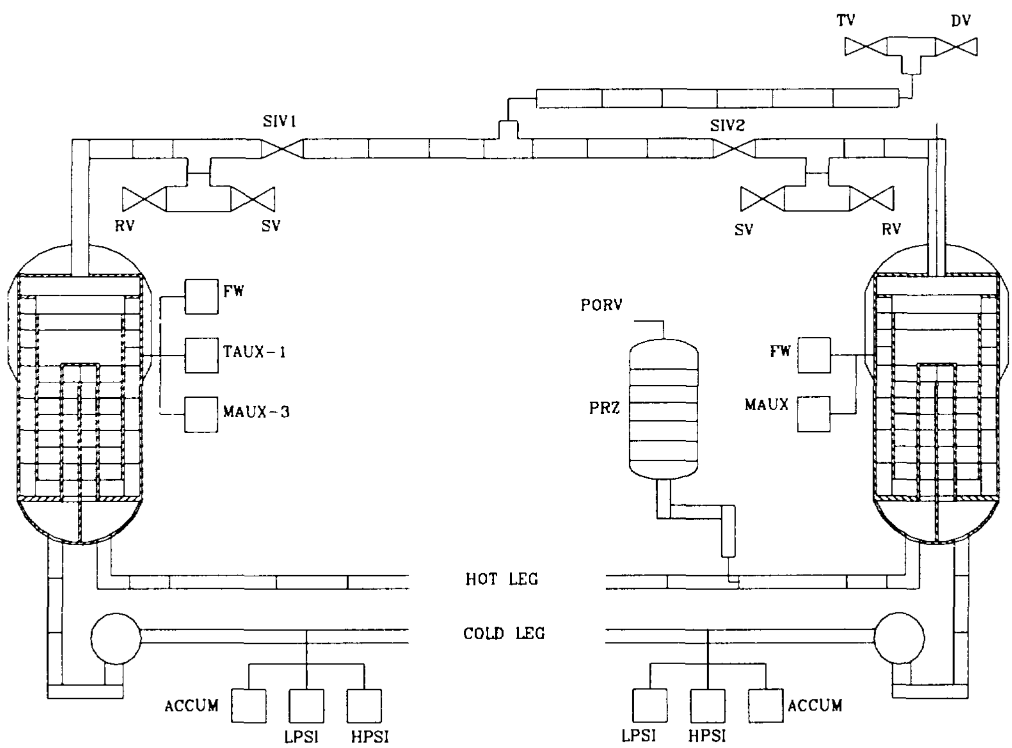

Figure 2. The whole SCDAP/RELAP5 model, except vessel, of Ringhals 2. The dubble loop steam generator is to the left and the single loop steam generator to the right.

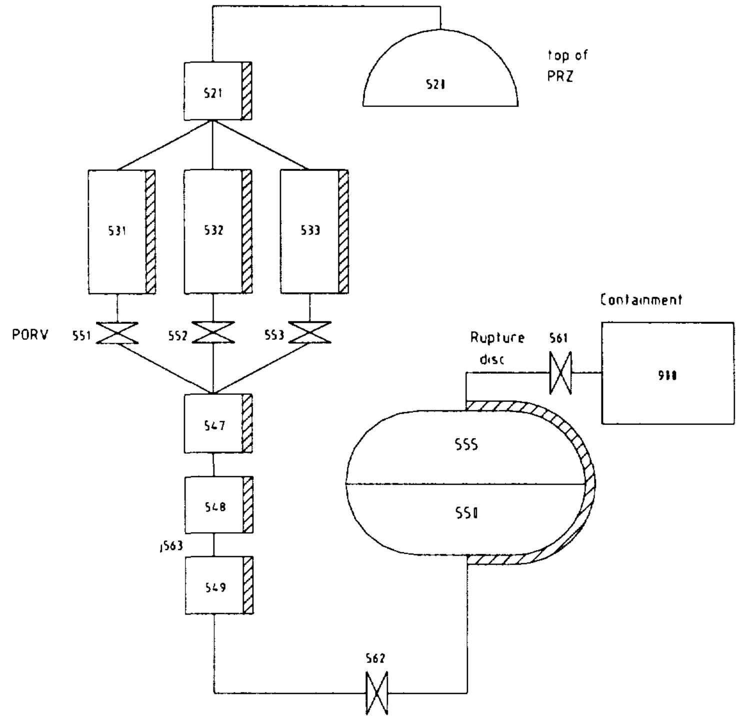

Figure 3. Nodalization scheme for power operated relief valves (PORV), pressurizer relief tank (PRT, volumes 555 and 550), and rupture disc from PRT to containment.

A detail of the nodalization of the power operated relief valves, PORV and the pressurizer relief tank, PRT is shown in Figure 3. Valve 562 is added to the model for stability reasons. Without it the water in the pressure relief tank, PRT, was splashing up through the pipes in an unphysical manner. It is modelled to open at the same time as the power operated relief valves are.

A radial nodalization of the core was done, according to Figure 4. The core was divided into five radial channels. The squares in Figure 4 each represent a fuel element. There are 157 fuel elements in the core of Ringhals 2. Each fuel element consists of a 15x15 grid with 204 fuel pins and 20 control rods.

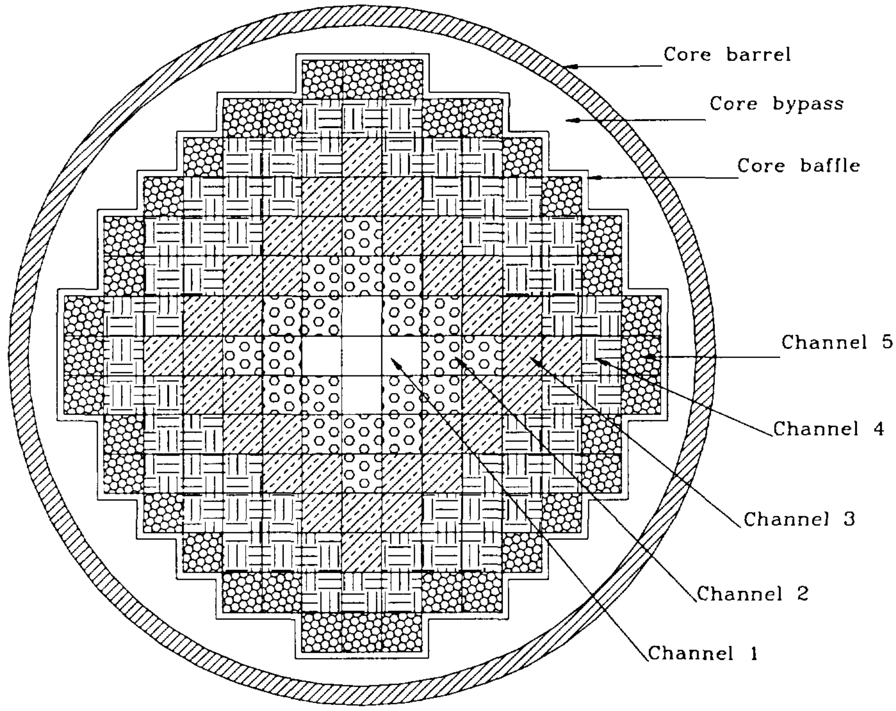

Figure 4. Radial nodalization of the core.

In SCDAP/RELAP5 the containment is modelled as one volume. It is the mass flow rate of fission products into this volume which is to be the input to the CONTAIN calculations. The containment volume is connected to the primary system through five valves, placed where one could expect a breach to occur. One connection is from the vessel bottom, and is to simulate the melt through of the vessel. A second connection is from the surge line, and is to simulate a surge line break. A third connection is to represent a melt through of the pipes leading to the power operated relief valves, at the top of the pressurizer. A fourth way for fission products to escape into the containment is through the rupture discs on the pressurizer relief tank. The fifth and last escape route is through the pressurizer safety valve.

The melt through of the vessel is modelled with the COUPLE creep rupture routine, and the surge line creep rupture and ruptures on the pipe leading to the power operated relief valves are modelled with the SCDAP creep rupture routine. The fission products in the primary system are modelled with the SCDAP/RELAP5 routine Fission product and aerosol transport.

3.2 Nodalization of vessel bottom

The vessel bottom was modelled in COUPLE. The axi-symmetrical nodalization is shown in Figure 5.

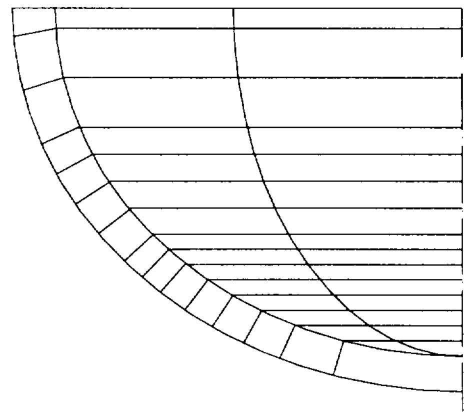

Figure 5. Vessel bottom nodalization for COUPLE.

3.3 Nodalization of the containment

The containment is modelled in CONTAIN1.2. The nodalization is shown in Figure 6. The figures represent the following parts of the containment:

1	Keyhole
2	Lower compartment
3	Annular compartment
4	Compartment for double loop steam generator
5	Compartment for single loop steam generator
6	Dome
7	Scrubber
8	Environment

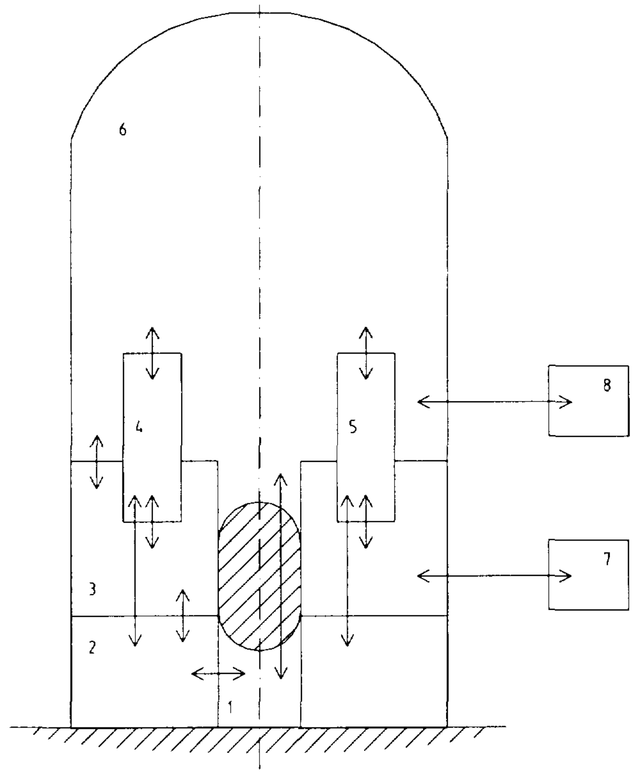

Figure 6. Nodalization of containment.

The arrows in Figure 6 represent junctions between the cells. Structures are modelled but not included in the figure. The rupture discs on the pressure relief tank and the surge line are placed in the annular compartment, cell three, and the safety valves on the pressurizer are placed in the dome, cell six.

In sequence one, see Section 4.1, there is a leakage from the dome to the environment, which is modelled with the junction between cells six and eight.

3.4 Initial conditions and set-points

When the transient starts the model plant is running at full power, with just a few days left before the beginning of the coast down. Full thermal power is 2 660 MW. Data used to model the core are collected from the actual core in use at Ringhals 2 in April 1997.

The initial conditions stated in Table 1 are the conditions calculated by the code during a steady-state run. The set-points in the same table are real conditions at Ringhals 2 during normal operation.

Table 1. Initial conditions vs. set-points.

	At end of steady-state	Set-points
Thermal power (MW)*	2660	2660
Pressure - pressurizer (Pa)*	1.55E7	1.56E7
Pressure - sg dome (Pa)	5.94E6	5.9E6
Temperature - hot leg (°C)	320.87	321
Temperature - cold leg (°C)	286.95	287
Main feedwater flow (kg/s)*	1415	1415
Main feedwater temperature (°C)*	211	211
Flow to turbine (kg/s)	1420	-
RC-flow (kg/s)	13891.9	13904

* Boundary conditions.

The difference between the feedwater flow and the flow to turbine are due to fluctuations in the flow out from the steam generator. Set-points for when the different valves are opened are found in Table 2.

Table 2. Opening pressures.

	Opening pressure (Pa)
Power operated relief valves-PRZ	161.9E5
Safety valves - PRZ	171.3E5
Rupture discs - PRT	7.91E5
Steam relief valves -sg	72.39E5
Safety valves - sg 1	75.1E5
2	76.1E5
3	77.2E5
4	78.2E5
5	78.9E5
6	78.9E5

A heat-loss from the primary system, of 1.5 MW during normal operation, is assumed, and in the model extracted from the vessel walls. The heat-loss is modelled to depend on the temperature difference between the primary system and the containment.

3.5 Initial core inventory of fission products

The five radial channels in the core were the basis for the calculation of the core inventory of fission products, which was calculated using the code ORIGEN2. From the radial power distribution in the core an average power for each channel was calculated. The history of each fuel element, from April 1997 and back through all the cycles the element had been active in the core, was recorded. Fuel elements with the same history, i.e. fuel elements that every year had been placed together in the same radial channel, were collected together and constituted one ORIGEN2-run. In all, 27 ORIGEN2-runs were made. The output from all runs were added up, and a total fission product inventory could be calculated, see Table 3. The fission product mass distribution over the radial channels was also calculated and used in the SCDAP input.

Of the species stated in Table 3, Ag and Sn were excluded, all the rest are used in the SCDAP/RELAP5 input file. The reason for this choice is the want to explore the capabilities of the code, and thus as many species as the code allowed to be used in the input file, were input.

Table 3. Core inventory when transient starts, calculated with ORIGEN2.

Species	Initial core inventory (g)
Ag	3.05E3
Ba	6.86E4
Cs	1.37E5
I	1.05E4
Kr	1.84E4
Ru	1.02E5
Sn	6.07E3
Sr	4.43E4
Te	2.40E4
Xe	2.35E5
Zr	1.67E5

SCDAP/RELAP5 has a model for calculating the increase of fission products in the gap. The initial gap inventory was assumed to be zero. The code also includes a fission product transport routine which has been used.

4 SEQUENCE

Initially, several sequences were planned, but as the problems with the code increased, in the end only one sequence was studied. The initial event for this sequence is a total loss of power and a malfunctioning turbine driven auxiliary feedwater pump. It was assumed that there was no leakage from the main recirculation pump seals in this case.

A leakage from the containment to the environment will be assumed to be present from the start of the calculations. The leakage's area is 5 cm^2. Containment spray in the containment will be initiated at an absolute pressure of 3.1 bar.

Table 4. Pre-set times for the sequence. Loss of power at time t = 0 s.

Time	Event
0 s	Pump trips
2 s	Scram completed
3 s	Turbine valve closes
5 s	Main feedwater supply stops
2 h	No battery power left

In Table 4, a time-table for pre-set trips is given. After two hours, when the batteries have run out of power, it will no longer be possible to operate neither the power operated relief valves nor the steam relief valves. They will stay in closed position for the rest of the sequence. Instead the safety valves will open. All valves toggle to maintain a constant pressure.

5 RESULTS

Only intermediate results will be presented below. No final results could be reached. The main problems are related to the SCDAP/RELAP5 code.

5.1 Computational results

The calculations presented below, start with a 400-second steady-state run, i.e. the transient starts at 400 s.

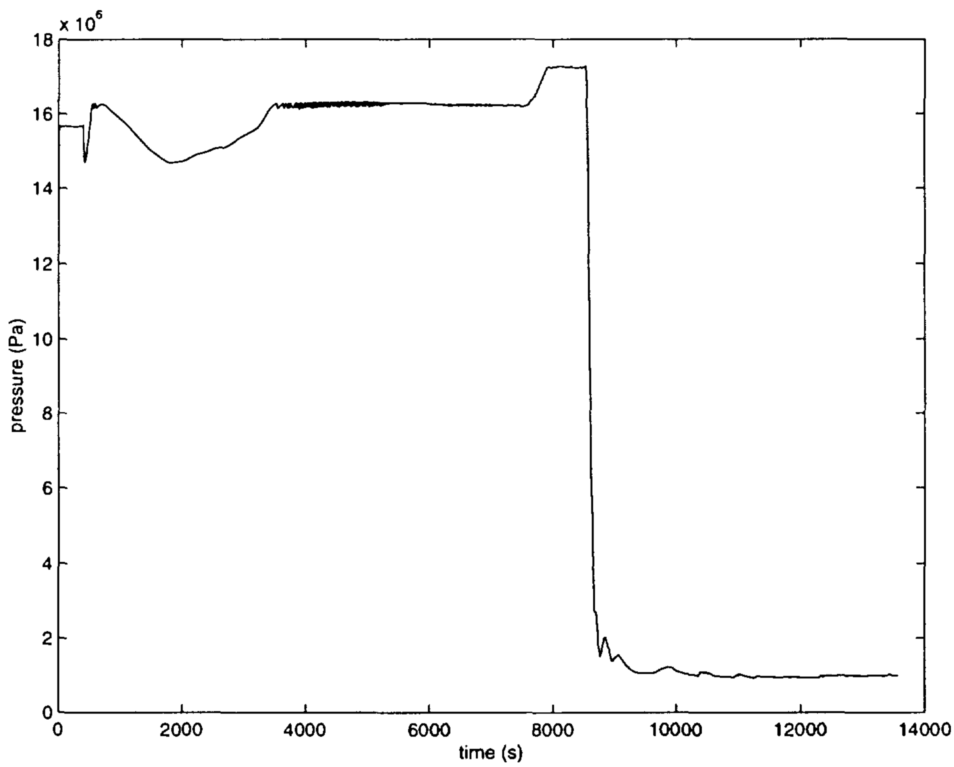

Figure 7. Pressure in the primary system.

After 400 s, when the loss of power occurs the pumps start to coast down immediately, see Table 4. The scram is assumed to be completed after 2 s. The pressure on the primary side initially decreases fast, se Figure 7. This is because the steam generators continue cooling the core during the first second after the scram, as one assumption was that the turbine valve would not close until 1 s after the scram was completed, Table 4. As soon as the turbine valve is closed the secondary side becomes a closed system and can no longer cool the primary system efficiently. The pressure on the primary side starts increasing and continues increasing until the power operated relief valves, placed in connection to the pressurizer, open. They open and close a few times, keeping the pressure constant.

While this happens on the primary side the pressure has continuously increased on the secondary side. When it reaches 72.39 bar the steam relief valves open, and the steam generators once again can cool the core, and the pressure on the primary side starts decreasing. The steam relief valves will open and close and try to keep the secondary side pressure constant. Approximately 30 min. after the transient has started the steam generators are dried out and there is no longer any cooling of the core. Subsequently the pressure on the primary side starts increasing again and continues increasing until the power operated relief valves open for the second time.

The power operated relief valves open and close to keep the pressure constant, visible in Figure 7, only through a thicker line here. Between 5000 and 6000 s these valves are forced to stay in an open position to prevent the pressure from starting increasing again, a thin line in Figure 7. This is because the water starts to boil, causing a massive steam generation.

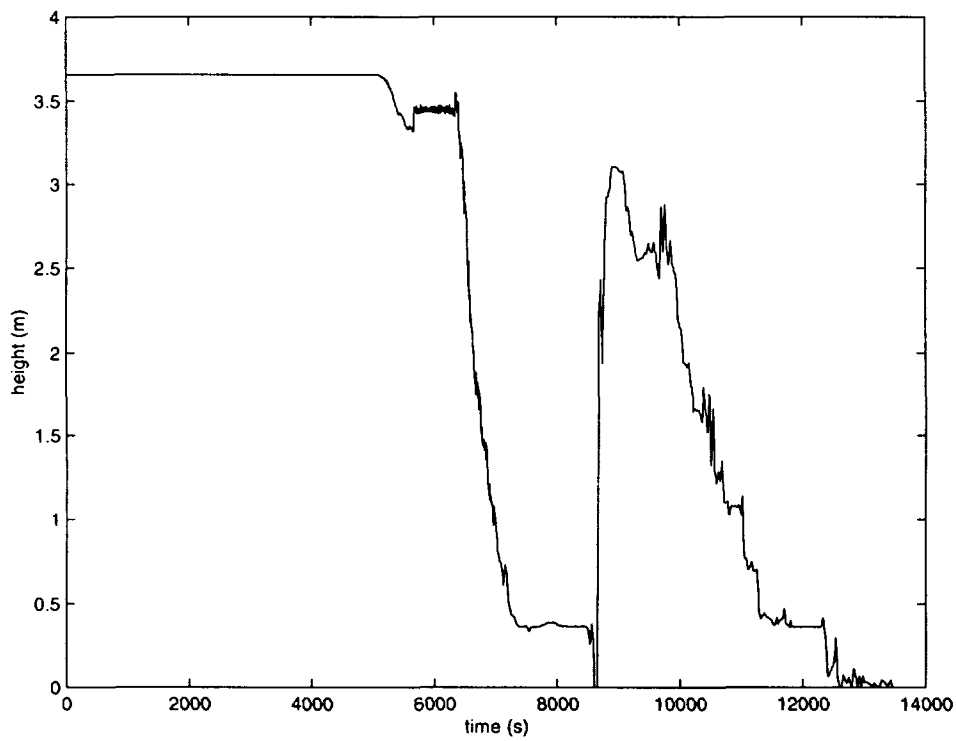

Figure 8. Collapsed liquid level in core.

When the core is getting uncovered, around 6000 s in Figure 8, the contact area between the fuel and the water is decreasing and less steam is generated. The power operated relief valves do no longer need to stay open all the time, but start toggling again.

Two hours after the transient started the batteries are assumed to run out of power, and the power operated relief valves can no longer be operated. They are assumed to close. The pressure on the primary side starts increasing again, until the pressure reaches 171.3 bar and the safety valve connected to the pressurizer opens. This valve also toggles keeping the pressure constant.

The gas temperature in the surge line starts increasing, see Figure 9. After 2 h and 16 min a creep rupture occurs in the first node of the surge line. The pressure immediately drops and the three passive accumulators are activated when the pressure drops below 49 bar in the cold leg, and empty their content of water into the primary system.

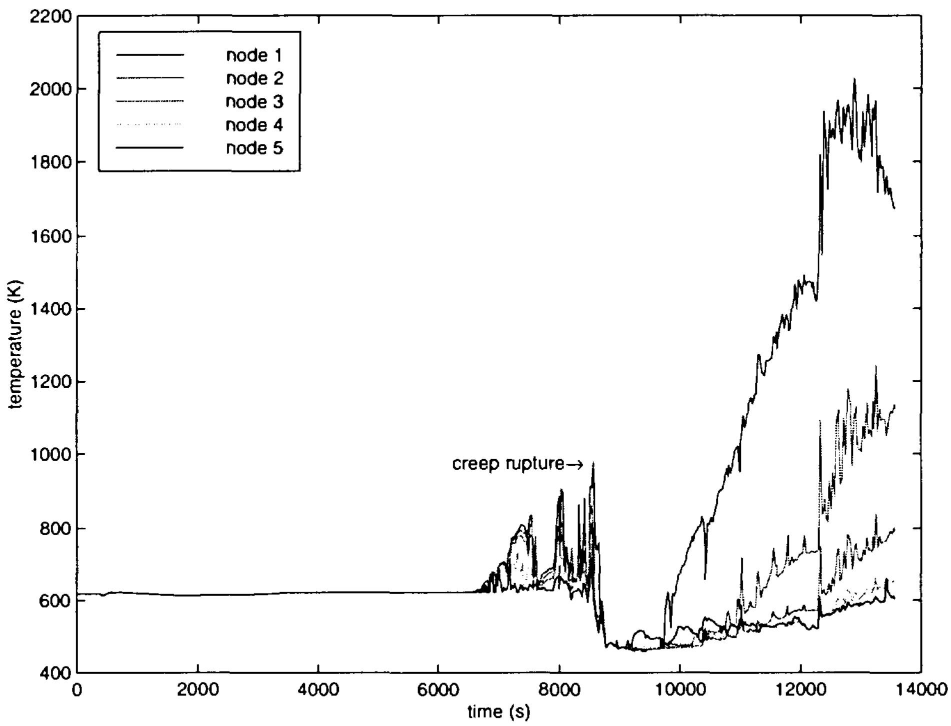

Figure 9. Gas temperature in surge line. Node one is the bottom node and node five the node closest to the pressurizer.

The water volume in one of the accumulators is plotted in Figure 10. The accumulators are emptied in 300 - 400 s.

The maximum surface temperature is plotted in Figure 11. This maximum temperature is identical to the top node, node 10, cladding temperature in Figure 12. The top node in the core is the node that will first become uncovered and where the heating up and also the melting start. As the core is uncovered the cladding temperature increases. After the surge line creep rupture, when the accumulators have emptied their water content into the primary system, the water level in the core increases fast, see Figure 8, and the cladding surface temperature immediately drops. Only the top node temperature remains high as the water level never reaches that far.

As the water level decreases again the temperatures in the core start increasing another time. The temperature increases until it reaches the melting point, the material in that node melts and the temperature drops. Output from SCDAP shows how the fuel is melting and whether any blockage occurs, see Figures 13 and 14.

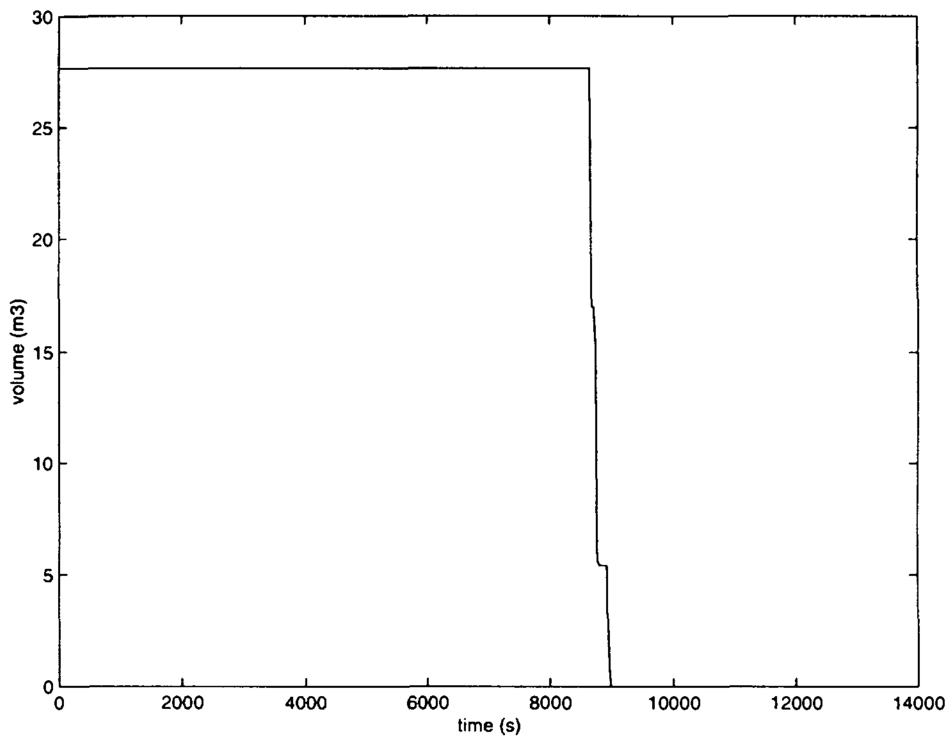

Figure 10. Water volume in one of the accumulators.

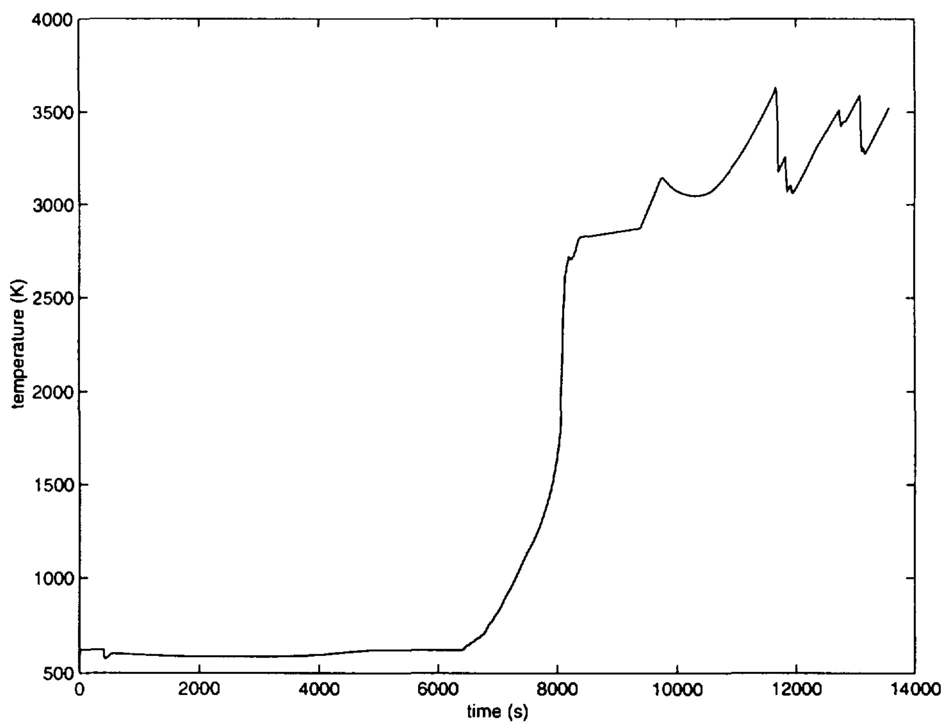

Figure 11. Core maximum surface temperature.

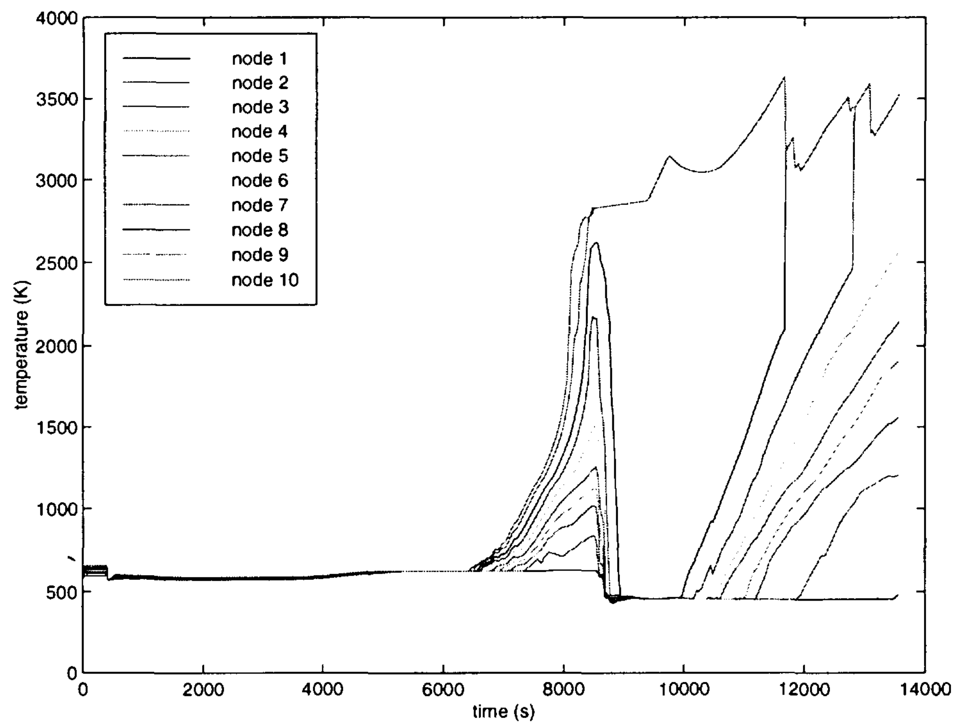

Figure 12. Outer cladding temperature of core centre channel.

core degradation map

I = intact fuel component
P = porous debris
M = molten or frozen ceramic pool
V = Relap fluid volume now void of fuel

underscore indicates metallic or planar blockage in volume at bottom of indicated node

__I__ __V__ x's indicate that total or bulk xxPxx blockage occurs xxMxx in the volume

Channel #	1	2	3	4	5
Axial node #					
10	V	I	I	V	I
9	xxMxx	xxMxx	xxMxx	xxMxx	I
8	P	P	P	P	I
7	I	I	I	I	I
6	I	I	I	I	I
5	I	I	I	I	I
4	I	I	I	I	I
3	I	I	I	I	I
2	I	I	I	I	I
1	I	I	I	I	I

Figure 13. Core degradation map after 2 hours and 17 minutes. Channel 1 is the centre channel.

core degradation map					
Fuel channel #	1	2	3	4	5
Axial node #					
10	V	V	V	V	I
9	V	V	V	V	P
8	xxMxx	xxMxx	xxMxx	xxMxx	P
7	P	P	P	P	I
6	I	I	I	I	I
5	I	I	I	I	I
4	I	I	I	I	I
3	I	I	I	I	I
2	I	I	I	I	I
1	I	I	I	I	I

Figure 14. Core degradation map, after 3 h and 30 minutes.

Table 5. Calculated times for major events.

Time	**Event**
2 min	PORV opens first time
30 min	Steam generators are dried out
52 min	PORV opens second time
1 h 40 min	Core starts getting uncovered
1 h 56 min	Core uncovered
2 h 5 min	Melt through of first guide tube
2 h 7 min	Failure of fuel pin cladding
2 h 16 min	Surge line creep rupture
-	Core slumping to lower head

Major events are summarized in Table 5.

The SCDAP/RELAP5 built-in routine for creep ruptures has been used. The creep rupture occurs at node one, at a temperature slightly below 1 000 K, after 2 h and 16 min. This is far below the melting temperature of the steel.

After 13 000 s the calculations started crashing over and over again, even when very small time steps were used, and the aimed goal to reach vessel melt through had to be given up.

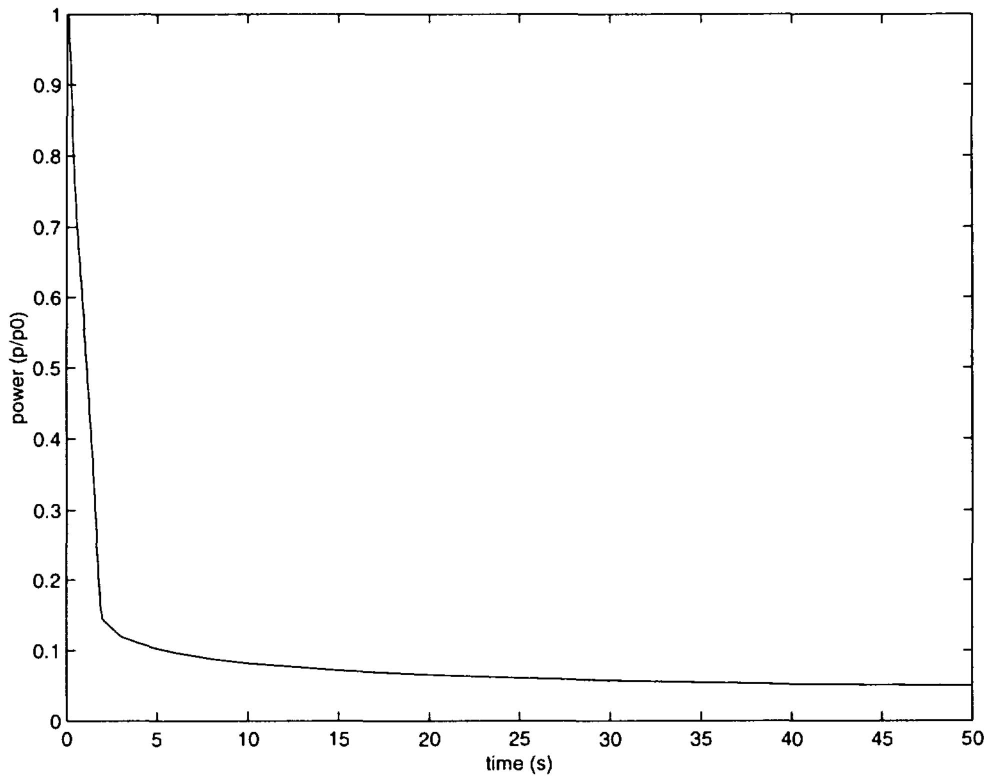

Figure 15. Power decay when transient has started. Power is plotted as ratio of full power.

5.2 Fission product output

The results needed for the final CONTAIN calculations, the calculations that would produce the final source term results, was detailed information about the fission products that did leak into the containment, i.e. where the leaks were situated and the leaked mass as a function of time, and this was meant to be calculated with SCDAP/RELAP5.

To get this information the SCDAP/RELAP5 routine Fission product and aerosol transport was used. It turned out that when this routine is activated it is not possible to make any changes to the component input of the model, at a restart. Desired changes include e.g. changing an opening time of a valve. The only changes discovered to be admitted at a restart were changes to the trip conditions. Moreover, the running times were extremely long. The critical events did not take place until after two hours, i.e. after 3 days to 1 week of running time on a Sun Ultra SPARC 20 work station.

The output from the code, concerning the fission products, is:

a Fuel fission product inventory
b Fuel rod gap inventory
c Release of noncondensibles to coolant
d Release of soluble fission products to coolant
e Fission product transport

a The initial fission product inventory, calculated with ORIGEN 2 and given in the input file, includes: Ba, Cs, I, Kr, Ru, Sr, Te, Xe and Zr, see Section 3.4. All these were accepted by the code, as can be seen in the input echo, where the code tells whether it has accepted the input or not and how the code has interpreted the given input. When later, the calculations begin, the output is reduced to comprehending only Xe, Kr, Cs, I and Te, i.e. four species that were input are ignored by the code.

b The fuel rod gap inventory is initially assumed to be zero, and is slowly being filled with Xe, Kr, Cs, I, He and H_2 during the calculation. When the fuel rod cladding is melt through, the code assumes the whole gap inventory to escape instantaneously into the coolant.

c and d The cumulative release of non-condensables is given as the sum of Xe, Kr, He and H_2 and the cumulative release of soluble fission products as the sum of CsI and CsOH. They are plotted in Figure 16.

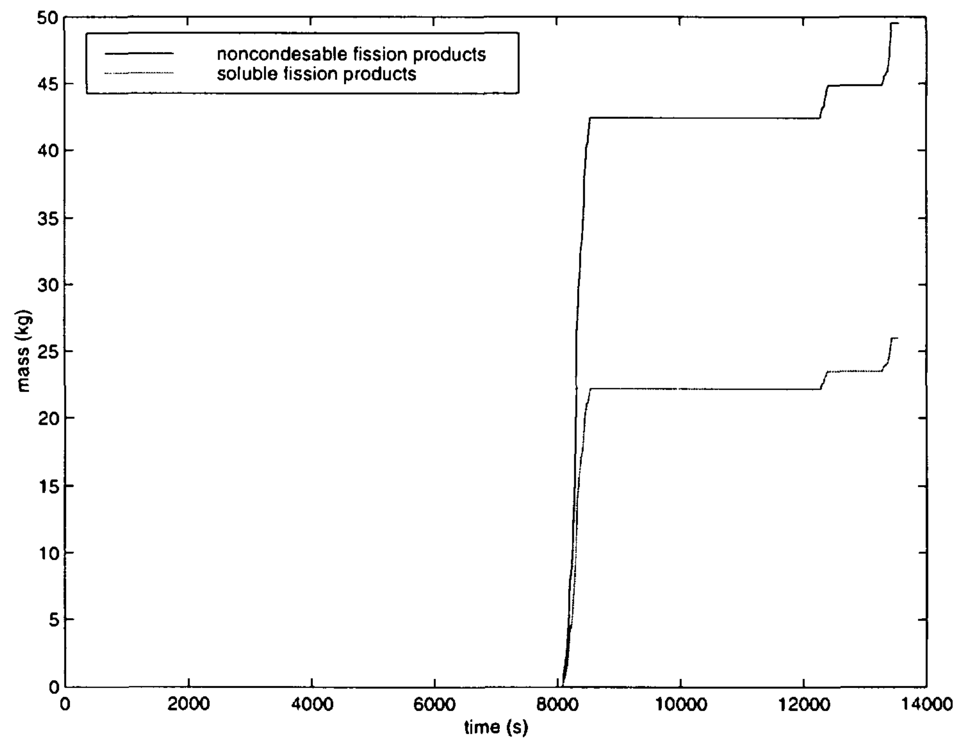

Figure 16. Core cumulative non-condensable and soluble fission product release.

e The fission product and aerosol transport routine is supposed to give the content of fission products in the hydrodynamic volumes. For each volume it gives information about how much fission product mass is carried by the liquid phase and how much is in the form of vapour and whether there is any fission product source in the volume and how much it produces in kg/s. The fission products condensed on heat transfer surfaces are also tracked, as well as fission products absorbed and deposited on heat structures. All this information is given for each species in each volume.

It is necessary to define in the input file which fission products and aerosols that are to be studied. According to the manual, [6] it is possible to study as many as 14 different species. Of these, ten were interesting in this case, but only seven of these ten species were accepted by the code. There is obviously an error in the manual. The seven species accepted by the code were in this case: I, CsI, CsOH, Te, HI, UO_2 and Ru.

Of the species released from the fuel, and presented in the output according to c) and d) above, none of the non-condensables can be further tracked.

It was discovered that the amount of CsI and CsOH released from the fuel according to output d) did not agree with the amount recorded by the transport routine, e). According to the cumulative release output the release of CsI and CsOH was of the order of 10 kg, see Figure 16, but according to the transport routine the primary system, including the containment, did only hold a mass of CsI and CsOH of the order of 10^{-17} kg, all phases counted.

5.3 Computations

The calculations became, for various reasons, very time consuming. As was mentioned in Section 5.2 when the Fission product and aerosol transport routine was activated, no changes to the component input were possible in the restart input file. It was only possible to change time step control cards and trip input data.

Short time steps were necessary in order to prevent the code from crashing. Normally a time step interval of 10^{-7} to $5 \cdot 10^{-2}$ was used, but it was not unusual that the interval 10^{-7} - 10^{-3} was necessary. These are very short time steps in relation to most time intervals of core degradation events or to the time a vessel melt through takes to develop.

After 13 000 s the calculations started crashing over and over again, even when very small time steps were used, and the aimed goal to reach vessel melt through had to be given up.

One test run was also made with the interim version SCDAP/RELAP5/Mod 3.2. The same contradictory results were obtained for the fission product transport as with the Mod3.1.8E version though. Also the MOD3.2 interim version tended to start crashing earlier than the Mod3.1.8E version. It started crashing already during the core uncovering, while the Mod3.1.8E version did not crash until far into the core melt phase.

6 CONCLUSION

The final goal of determining the source term for Ringhals 2 could not be fulfilled, mainly due to problems related to the SCDAP/RELAP5 code. The execution times for this code are extremely long which violates its practical utilization, especially for source term calculations at which the restart option is much limited.

However, substantial results were obtained in the project, producing complete and quality controlled input models for SCDAP/RELAP5 and CONTAIN. Moreover, one selected severe accident scenario was simulated, a total station blackout sequence, which was calculated with SCDAP/RELAP5/Mod3.1E. Also, some tentative calculations were made with CONTAIN1.2 using guessed values of fission product sources in order to test the containment model.

At the simulation of the total station blackout with SCDAP/RELAP5 loss of power was assumed to occur at time zero. Other assumptions made were, e.g. that the scram takes 2 s to be completed and that the turbine valve is closed after 3 s. These intitial times were varied and affect the initial development but are less important in the long term aspect. The steam generators were emptied about 0.5 h after the initial loss of power. After about 1 h the pressure on the primary side had increased enough for the power operated relief valves to open. After 2 h these valves were assumed not to be operable any longer, and the safety valve assumed to open in stead. After 2.25 h a surge line creep rupture occurred. The primary system was immediately depressurized. The calculations started crashing repeatedly and vessel melt through was never reached.

The task will now halt until another release of SCDAP/RELAP5 with improved capabilities for source term calculations and improved algorithm for faster calculations is obtained.

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