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Station Blackout Analysis of Nuclear Power Plant Using Source Term Code Package

Analiza nezgode z izgubo napajanja v jedrski elektrarni z uporabo STCP

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Abstract

Continuous efforts to ensure the safety of nuclear installations in Slovenia have led to comprehensive analysis of Levels II and III of hypothetic station blackout accident modelled using the tools at our disposal. This paper represents the thermal hydraulic and radionuclide transport part of the overall effort.

MARCH3 and VANESA modules of Source Term Code Package were used to analyze four different scenaria depending on different reactor coolant pump leak rate (125 gpm and 400 gpm, respectively) and containment design pressure (i.e. 0.309 MPa and 0.785 MPa). The final aim of the project was to prepare input into the Level III analyses of the accident.

The accident starts by loss of off-site power combined with loss of diesel generators. The turbine driven auxiliary feedwater pump operates additional two hours after the inception of the accident. The results are given in form of graphs displaying reactor coolant system and containment parameters.

Povzetek

Napori, usmerjeni v varno obratovanja jedrskih naprav v Sloveniji, vodijo v smeri varnostnih analiz nivoja II in III, kamor spada tudi analiza nezgode z izgubo napajanja. V pričujočem prispevku so obdelani termohidravlični aspekti in aspekti transporta radionuklidov.

Z moduloma MARCH3 in VANESA smo obdelali štiri različne scenarije, odvisni od količine puščanja reaktorske črpalke (125 in 400 gpm) ter tlaka porušitve zadrževalnega hrama (0.309 in 0.785 MPa). Končni namen projekta je bila priprava vhodnih podatkov za analizo nivoja III izbrane nezgode.

Nezgoda se prične z izgubo zunanjega napajanja in izgubo dieselskih generatorjev. Turbinska črpalka deluje še dve uri po pričetku nesreče. Rezultati so prikazani v obliki grafov parametrov primarnega kroga in zadrževalnega hrama.

Introduction

Loss of power is one of the most dominant sequences arising through Level I Probabilistic Safety Assessment and was therefore chosen for analysis using available Source Term Code Package (STCP).

STCP is older albeit still widely used code package utilized for severe accident analysis.

Based on actual plant geometry it offers assessment of source term as well as various thermal hydraulic phenomena. It is both robust and modular and can be applied for analysis of most western PWR and BWR designs.

Two modules, MARCH3 (thermal hydraulics) and VANESA (radionuclides generation and distribution inside containment) have been used. Two loop nuclear power plant was modelled and studied. Main characteristics of the model have already been described by Marn et al (1993). Below, scenario and results are described and commented.

Scenario

After AC is lost the reactor pumps are stopped and reactor scrams. Diesel generators fail to start. The primary concern is to remove decay heat from reactor core, preferably through the steam generators whose levels decrease significantly due to bubble collapse. The main feedwater is stopped and replaced with auxiliary feedwater system using turbine driven pump. If turbine driven pump fails to start this means boil-off of water contained on the secondary side of steam generators followed by increase of core temperature and possible (unless other precautionary actions are taken by plant personnel) core degradation and meltdown.

This scenario assumes that the turbine driven pump starts successfully. After 30 minutes the instrument air used to control the pump is lost and the operator starts to control the turbine driven pump manually. After additional two hours the battery power is lost which results in loss of instrumentation and assumption of turbine driven pump failure.

Additional assumption is reactor pump seal leak which develops 30 minutes after the start of accident due to insufficient cooling of the seal as the component cooling system ceases to operate. The seal leak amounts to small break LOCA outflow of primary coolant. Two different values are proposed, namely 125 and 400 gpm, following the NRC course on Severe Accidents propositions.

There is no structural models imbedded in the STCP thus simple failure of containment is assumed after the pressure has reached predetermined threshold level of 0.309 (design pressure) and 0.785 (result of available IPE analysis) MPa. There are no further significant assumptions proposed.

It is presumed that after the turbine driven pump is stopped the amount of water leaking through seal leak does not suffice to dissipate decay heat, therefore the primary coolant starts to boil and uncovers the core. Amount of decay heat dissipated by uncovered core leads to its meltdown, reactor pressure vessel failure, and ultimately containment failure. The results of the analysis should give an operator estimate of amount of time available to take appropriate actions preventing the breach of containment integrity.

Results

Four cases, depending on amount of seal leakage and containment design pressure, were run. Table 1 shows results for reactor pressure vessel failure and containment failure, respectively.

Figures 1 through 5 show various parameters from the beginning of the accident until just after the reactor pressure vessel failure.

Figures 6 through 13 show various parameters from the beginning of the accident until the containment failure.

Table 2 shows the inventory released into the containment at the postulated time of containment failure.

Table 1. Time to failure.

	Time to reactor vessel failure [hr]	Time to containment failure [hr]
125 gpm & .309 MPa	6.662	6.687
125 gpm & .785 MPa	6.662	27.401
400 gpm & .309 MPa	3.761	3.768
400 gpm & .785 MPa	3.761	20.498

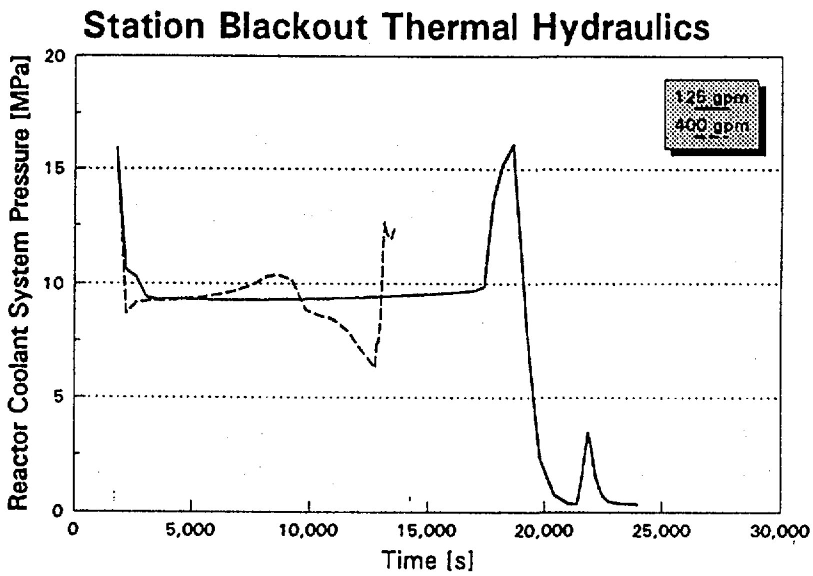

Figure 1. Reactor Coolant System Pressure.

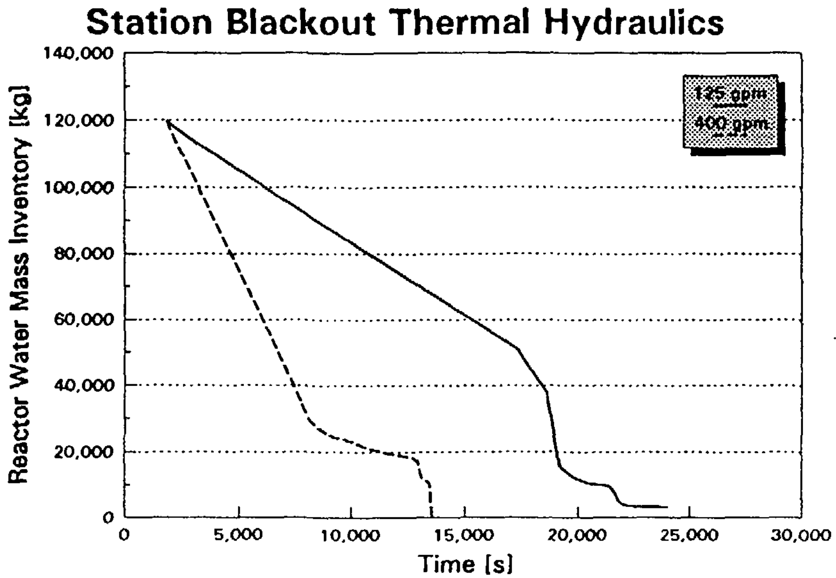

Figure 2. RCS Water Mass Inventory.

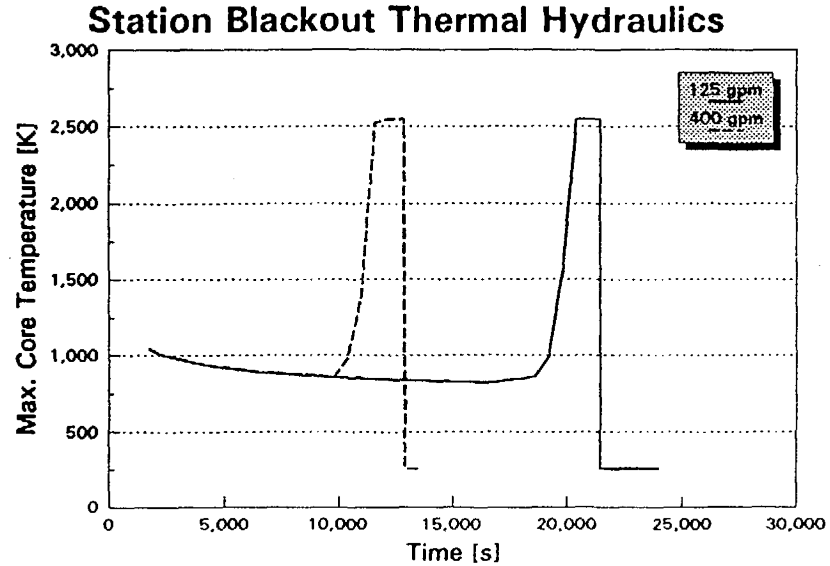

Figure 3. Max. Core Temperature.

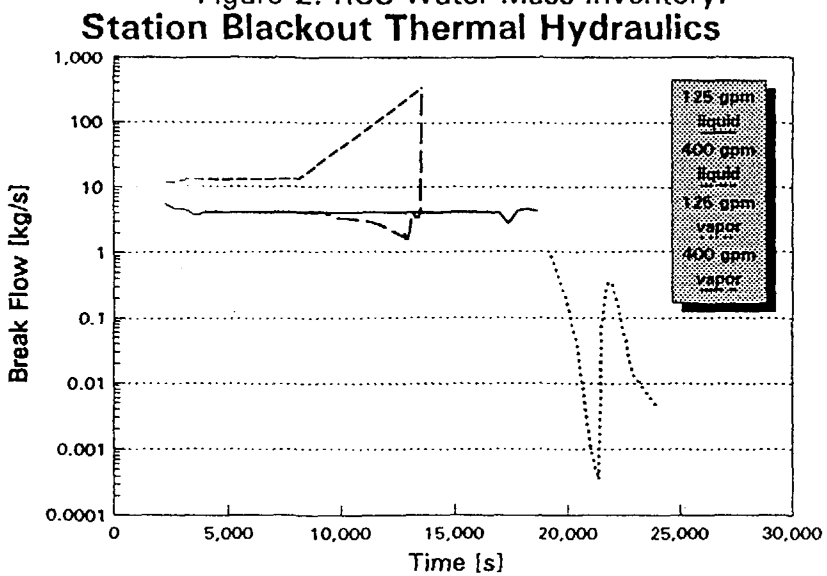

Figure 4. Break Flow Rate.

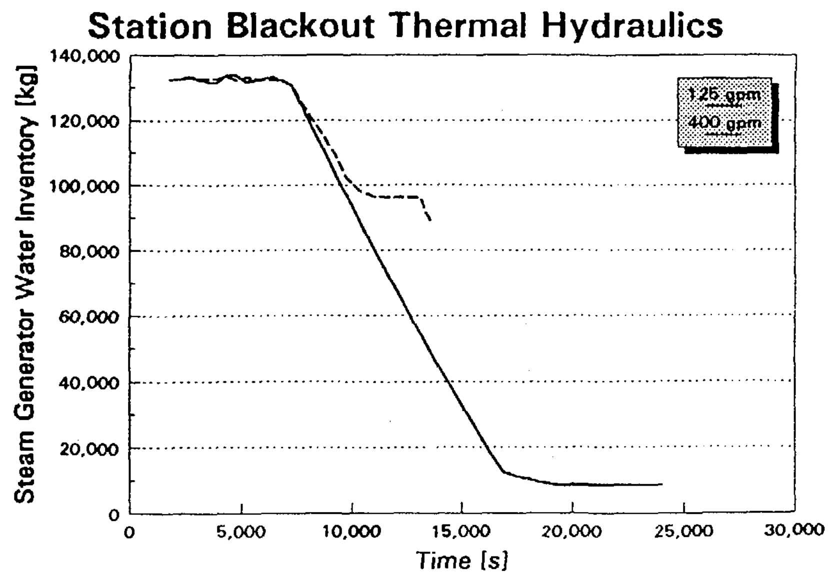

Figure 5. SG Water Inventory.

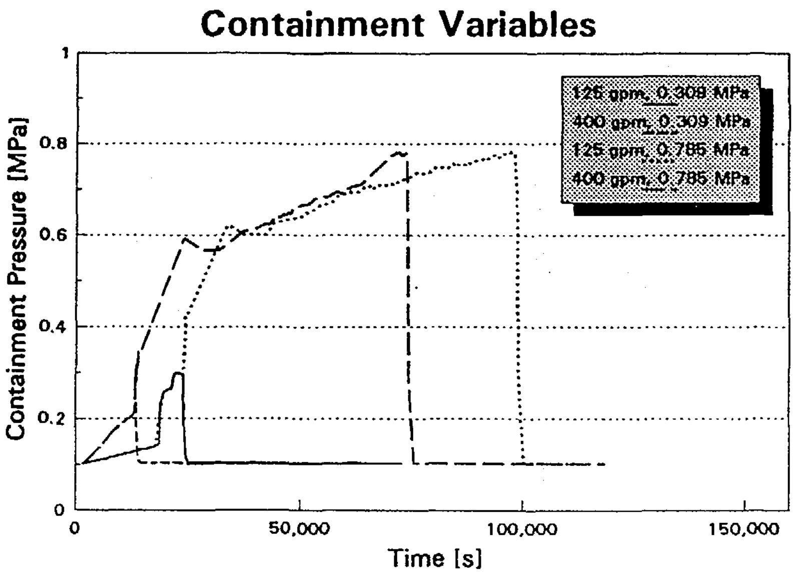

Figure 6. Containment Pressure.

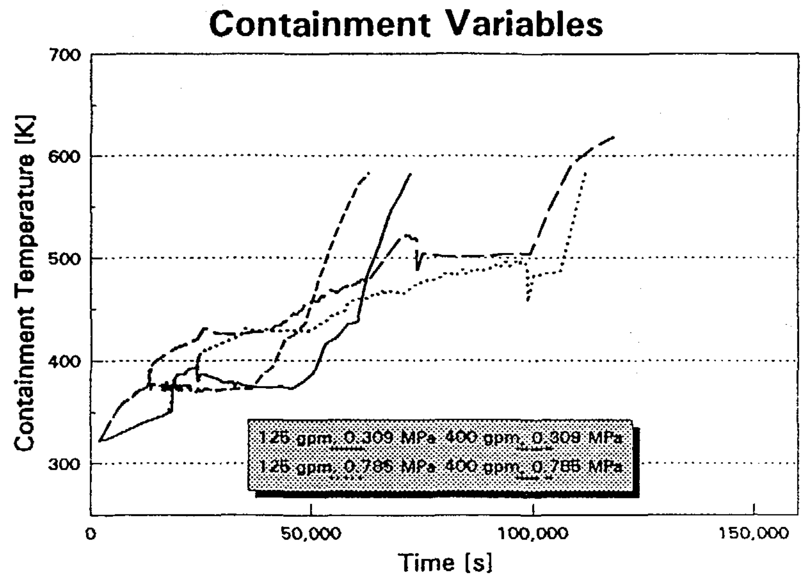

Figure 7. Containment Temperature.

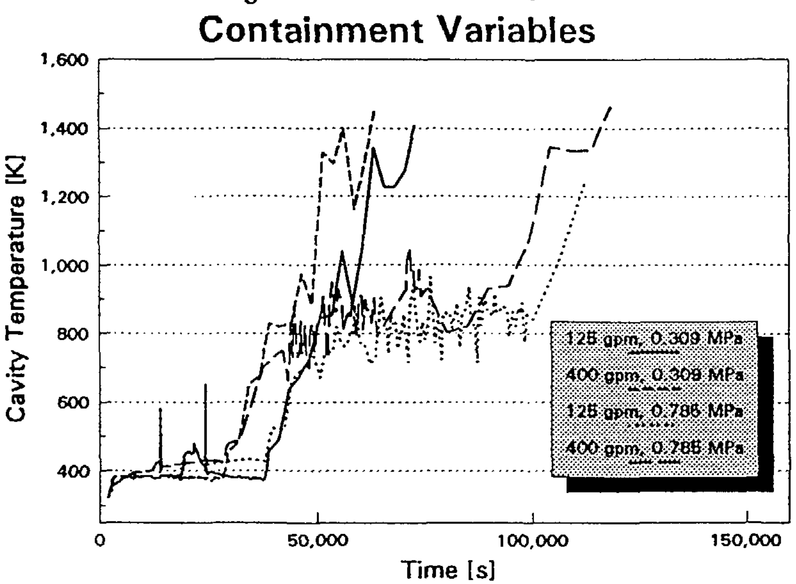

Figure 8. Cavity Temperature.

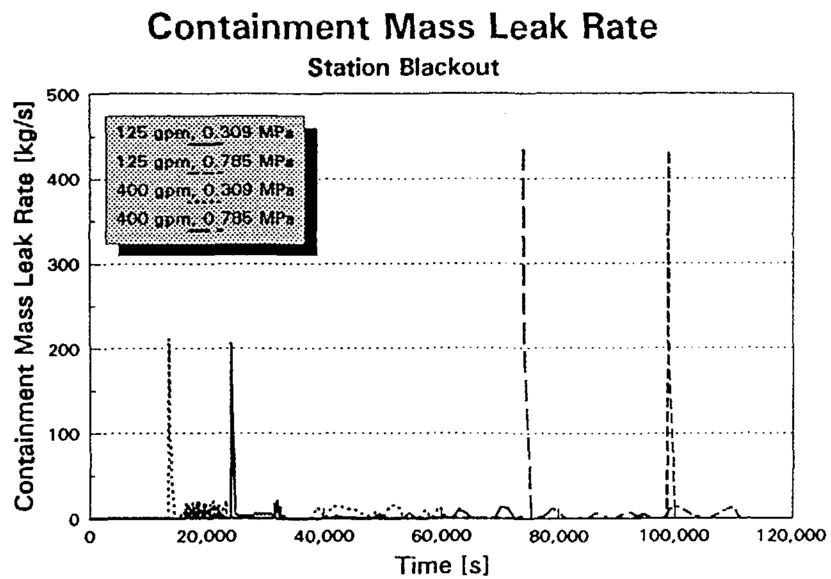

Figure 9. Containment Mass Leak Rate.

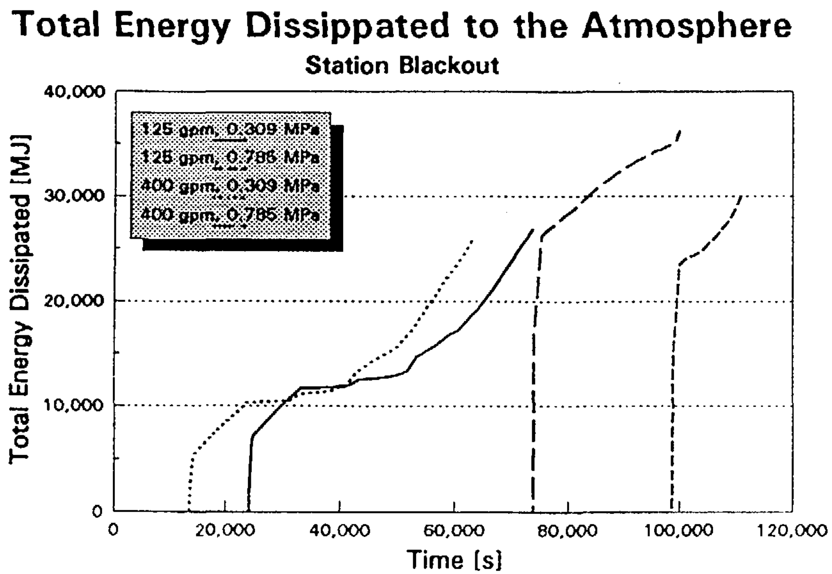

Figure 10. Total Dissipated Energy.

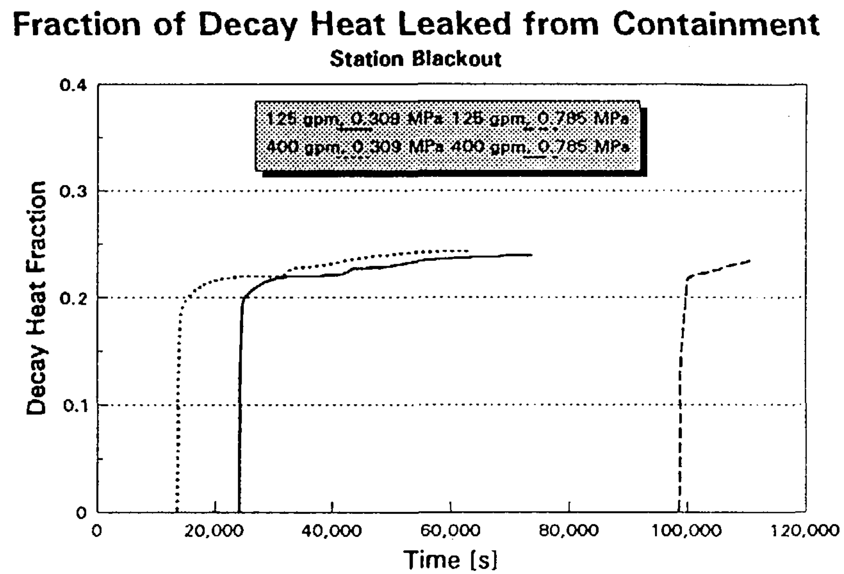

Figure 11. Fraction of Decay Heat Leaked.

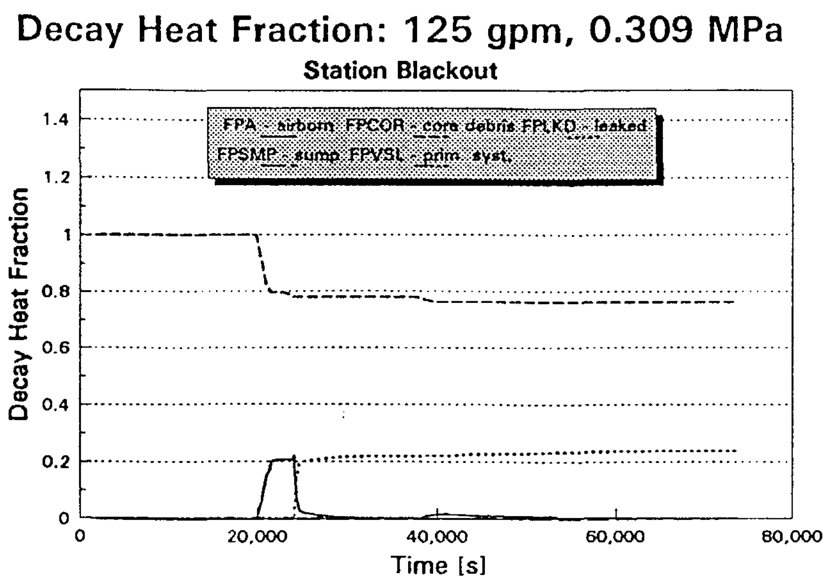

Figure 12. An Example of Temporal Evolution.

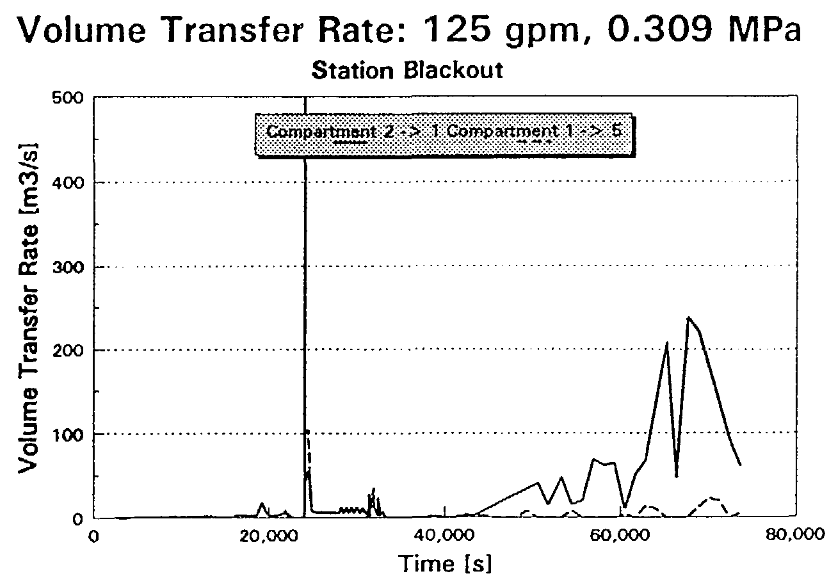

Figure 13. Volume Transfer Rate.

Table 2. Fraction of Inventory Released into Containment.

Case / Isotope groups	125 gpm and .309 MPa	125 gpm and .785 MPa	400 gpm and .309 MPa	400 gpm and .785 MPa
Noble gases	0.9163	0.9040	0.8960	0.8908
I group	1.0	1.0	1.0	1.0
Cs group	1.0	1.0	1.0	1.0
Te group	0.5628	0.4869	0.5466	0.5498
Ru group	$2.25x10^{-6}$	$2.07x10^{-6}$	$1.77x10^{-6}$	$2.41x10^{-6}$
Sr group	0.4442	0.2510	0.3020	0.2472
La group	$2.30x10^{-2}$	$2.61x10^{-3}$	$4.45x10^{-3}$	$2.67x10^{-3}$
Ce group	$4.41x10^{-2}$	$6.56x10^{-3}$	$1.03x10^{-2}$	$6.58x10^{-3}$
Ba group	0.2346	0.1787	0.1896	0.1705

Conclusion

The analysis shows that two distinct mechanism are responsible for the core damage. At 125 gpm the system does not cool itself enough (i.e. small break LOCA) which is characterised by relatively high pressure and vessel failure after steam generator secondary coolant boil-off. At 400 gpm, on the other hand, the leak suffices for decay heat removal (i.e. medium break LOCA) thus lowering pressure and causing vessel failure even before the steam generators were empty. Both mechanisms prompt different approaches to accident mitigation. Former requires sufficient amount of auxiliary feedwater while latter additional amount of water introduced into reactor coolant system.

Both sequences, however, result in fairly similar behavior of system parameters within the containment and same order of magnitude of radionuclides amount in containment atmosphere. The best results given the assumed scenario are at 125 gpm leakage and 0.785 MPa which is also intuitively correct.

Acknowledgment

The authors wish to thank Uprava R. Slovenije za jedrsko varnost for sponsoring the research.

Literature

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[2] Mavko, B., Marn, J., Leskovar, M., Kožuh, M., "Station Blackout Accident Scenario Analysis Using Code Package STCP", IJS-DP-7103, 1994.

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