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TMI-2 Analysis[1]

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The accident at Three Mile Island Unit 2 (TMI-2) provides an opportunity to benchmark severe accident analysis methods against full-scale, integrated facility data. In collaboration with the U.S. Department of Energy (DOE), the OECD Nuclear Energy Agency established a joint task group to analyze various periods of the accident and benchmark the relevant severe accident codes. In this paper the author presents one result from the TMI-2 analysis exercise that may be of interest in evaluating thermal hydraulics codes.

Arguably one of the more interesting aspects of the analysis results is the predicted pressurizer level response during core heatup. A brief review of the accident scenario is relevant to understanding the pressurizer behavior and the calculations[2]. Feedwater and turbine trips initiated the accident upon loss of condensate suction. When the pilot operated relief valve (PORV) opened in response to the reactor coolant system (RCS) pressure increase caused by the loss of heat sink it stuck open. The operators did not notice the stuck open PORV, but noticed the full pressurizer. The operators terminated high pressure injection and initiated letdown flow. The net loss of coolant through the PORV and letdown system caused the RCS coolant inventory to decrease to the point at which the RCS pumps could not be operated. By ~100 min the operators had tripped all four RCS coolant pumps, and core heat-up started at ~110 min. At ~140 min the operators closed the PORV block valve terminating the loss of coolant through the PORV. However, flow through the letdown system continued. The operators restarted the 2B RCS pump at 174 min for ~1 min. The restart pumped about 30 m^3 of liquid into the reactor vessel. RCS pressure increased rapidly during the 2B pump transient.

The above scenario is summarized in Figure 1 showing the measured primary and secondary system pressures. At ~30 min into the accident RCS pressure had decreased to secondary saturation pressure, and continued to follow the A-loop secondary pressure until about 125 min. At ~125 min RCS pressure started to increase departing from the A-loop once through steam generator (OTSG) secondary side pressure. The author infers that energy was no longer being transported to the A loop OTSG after about 125 min. Hydrogen generation due to cladding oxidation probably caused the loss of energy transport to the A-loop OTSG. RCS pressure continued to increase until the end of the 2B pump transient.

The measured pressurizer level and RCS pressure for 100 to 200 min after turbine trip are shown in Figure 2. After the last RCS pump trip the pressurizer level decreased following the decreasing RCS pressure. When the operators closed the PORV block valve the pressurizer level stabilized at a constant level. In the TMI-2 plant

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DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED a U-tube surgeline as shown in Figure 3 connects the pressurizer to the hot leg. The U-tube arrangement could allow the pressurizer and surgeline to act a manometer responding to the pressure differential between the hot leg and pressurizer dome. If hot leg pressure is above the pressurizer dome pressure, then some liquid will be retained in the pressurizer. Increasing RCS pressure at PORV block valve closure trapped the coolant inventory in the pressurizer.

The TMI-2 analysis exercise participants[3] found that under some conditions the severe accident analytical methods predicted that a severe accident might not have occurred due to pressurizer drainage after PORV block valve closure. Predicted drainage of the pressurizer cools the core delaying core heat-up. Codes such as SCDAP/RELAP5 predicted drainage depending on make-up flow rate, see Figure 4. In effect a bifurcation in pressurizer response is predicted. If liquid is converted to vapor (steam and/or hydrogen) in the core at a rate that is greater than the condensation rate, then RCS pressure increases, and the pressurizer inventory remains constant or increases. In this paper the author presents a sensitivity analysis of the factors that may influence the calculated pressurizer level after PORV block valve closure.

The phenomena considered include hydrogen generation rates, primary to secondary heat transfer, and energy generation in the core. The sensitivity analysis was conducted using RELAP5, thus allowing improved control of the parameters of interest. Figure 5 shows the SCDAP/RELAP5 compared to the RELAP5 calculations for a make-up flow rate (MUF) of 0 kg/s. At a MUF= 0 kg/s RELAP5 predicts drainage of the pressurizer, while SCDAP/RELAP5 predicts inventory hold-up in the pressurizer. The RELAP5 calculation for MUF= 0 kg/s is the base case for these sensitivity studies.

Parameter variations were generally initiated at 120 min. For the cases with hydrogen injection the injection rate was assumed to be a constant at either 0.01, 0.05, 0.1, or 0.5 kg/s. An injection rate of 0.1 kg/s is about equal to the average hydrogen generation rate for the SCDAP/RELAP5 base case. Another set of variations removed primary to secondary heat transfer at 120 min. I considered three cases: (a) nominal estimation of heat transfer before OTSG removal, (b) ten percent reduction in primary to secondary heat transfer before OTSG removal, and (c) 1/3 reduction in primary to secondary heat transfer before OTSG removal.

The effect of hydrogen injection is summarized in Figure 6. For the hydrogen injection rates of 0.01 and 0.05 kg/s do not lead to retention of the pressurizer liquid inventory. Pressurizer liquid inventory is retained for an injection rate of 0.5 kg/s (~5 times the SCDAP/RELAP5 calculated hydrogen generation rate). At an injection rate of 0.1 kg/s it is not clear that inventory will or will not be retained since RELAP5 aborted.

Another comparison is to the case where 25% additional decay heat, and hydrogen to simulate cladding oxidation heat. The calculated pressurizer level is shown in Figure 7 for this case, 0.1 kg/s hydrogen flow, and the SCDAP/RELAP5 base case. The calculated pressurizer level is about the same in all three cases. It may be concluded that hydrogen generation, and oxidation energy plays a direct roll in pressurizer level response.

The author also considered the primary to secondary heat transfer rate as one of the parametric sets. All primary to secondary heat transfer through the steam generator tubes was removed from the base case calculation at 120 min. For the parametric cases the heat transfer rate before OTSG heat transfer removal was reduced by 10% and 33%. These calculations are summarized in Figure 8. All three calculations indicate pressurizer drainage before 174 min. However, decreasing primary to secondary heat transfer does increase the time to the onset of drainage.

The calculated pressurizer response is a result of the pressure difference between the hot leg and pressurizer dome. In figures 9 and 10 the differential pressure (hot leg minus pressurizer) and pressurizer level for 0.1 kg/s hydrogen injection and OTSGs removed are compared. The author found that the pressurizer drained when the pressure differential fell below about 0.04 MPa (~6 psid). This occurred in all cases where RCS pressure did not increase or could not be sustained to ~174 min. For example, Figure 11 shows the calculated RCS pressure for a number of cases. Only in the cases with sustained increasing RCS pressure after 140 min are associated with pressurizer inventory retention.

A number of conclusions may be drawn from the predictions of pressurizer level. First the predicted pressurizer level after PORV block valve closure is determined by thermal hydraulics as well as the calculation of hydrogen generation. Second it is possible to arrive at a nonconservative prediction of pressurizer level (pressurizer drainage). Although some aspects of severe accidents may not depend greatly on thermal hydraulics, it is the author's conclusion that predictions of severe accidents require severe accident models properly coupled to reliable thermal hydraulics models.

REFERENCES

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[1] This work was supported by the U.S. Department of Energy (DOE), Assistant Secretary for Nuclear Energy, Office of Light Water Reactor Safety and Technology, under DOE contract DE-AC07-76ID01570.

[2] D. W. Golden, et. al., "Summary of the Three Mile Island Unit 2 Analysis Exercise," *Nuclear Technology*, Vol. 87, August 1989.

[3] TMI-2 Joint Task Group, *TMI-2 Analysis Exercise*, to be published.

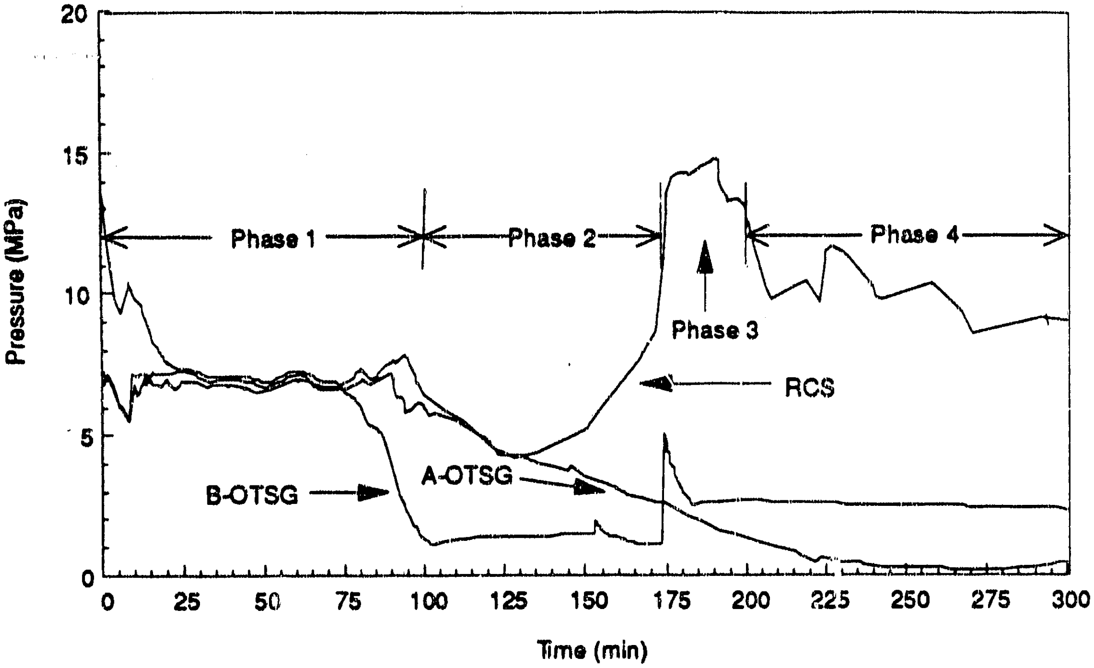

Figure 1. TMI-2 measured system pressures.

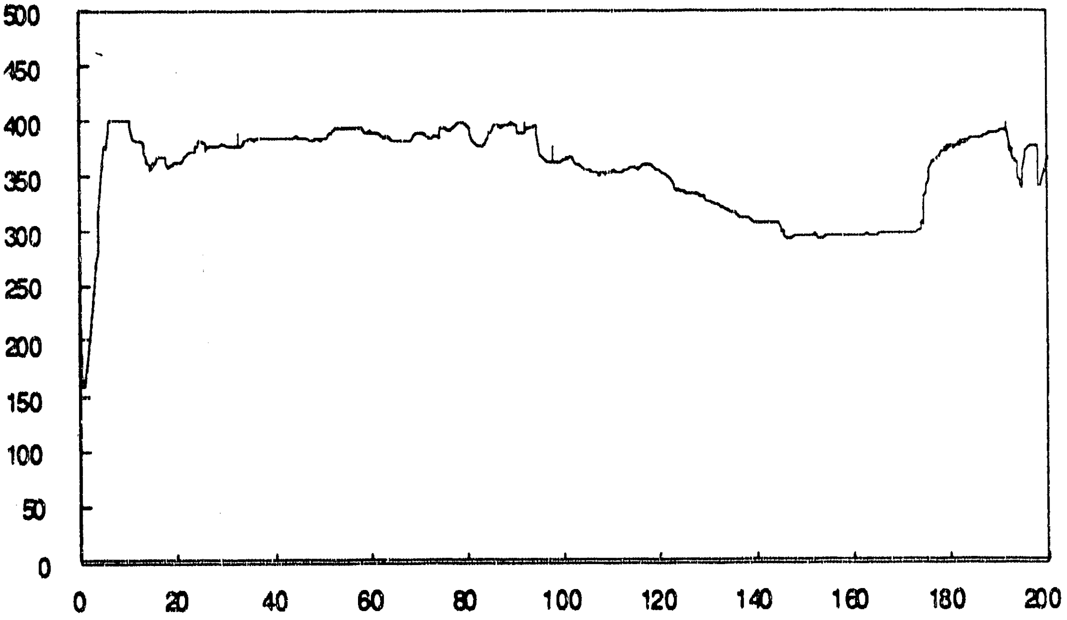

Figure 2. TMI-2 measured pressurizer level.

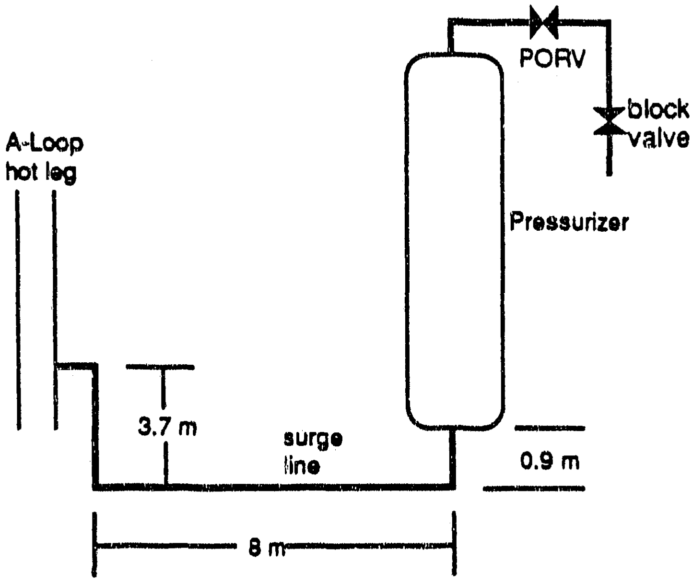

Figure 3. Pressurizer arrangement.

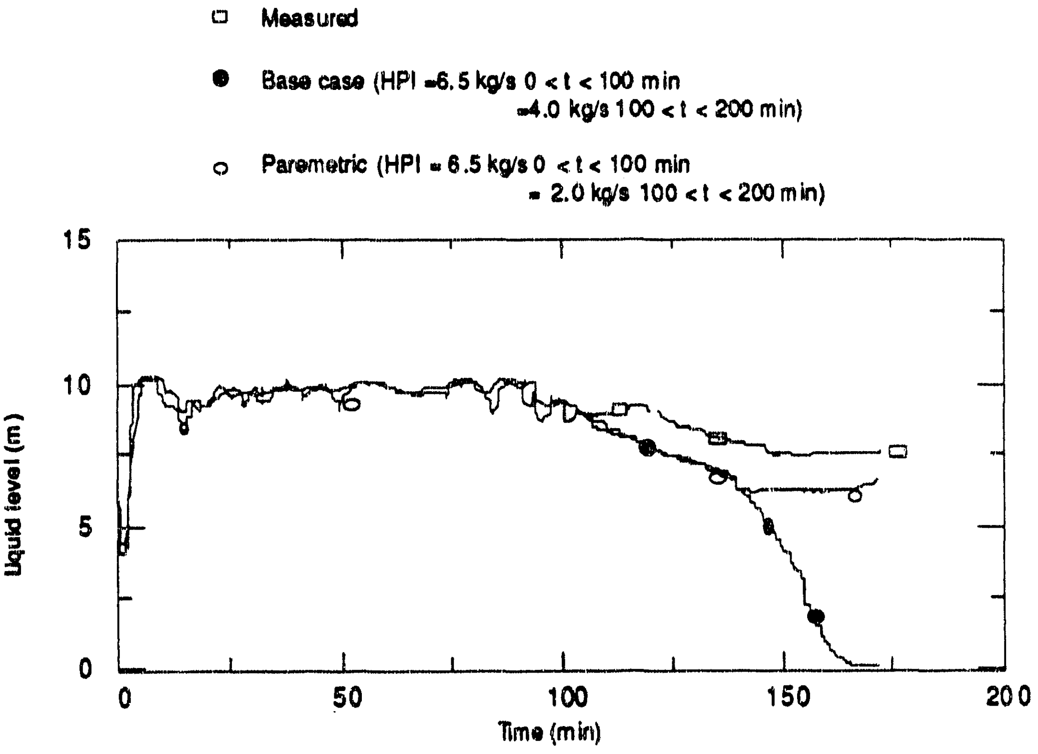

Figure 4. SCDAP/RELAP5 calculated pressurizer level vs. measured level.

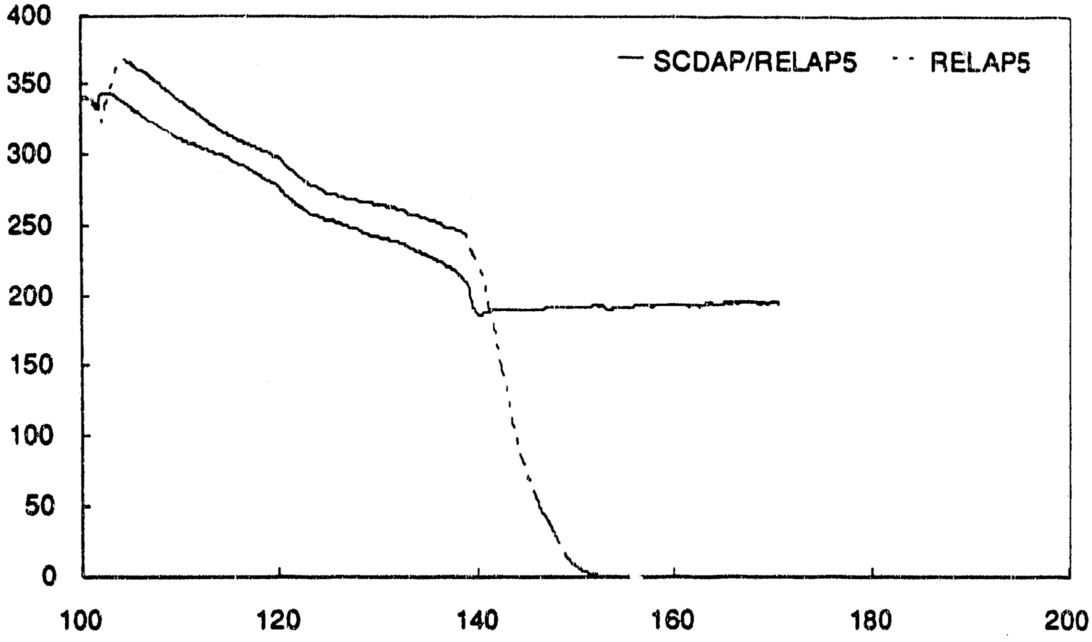

Figure 5. Comparison of SCDAP/RELAP5 and RELAP5 calculations of pressurizer level.

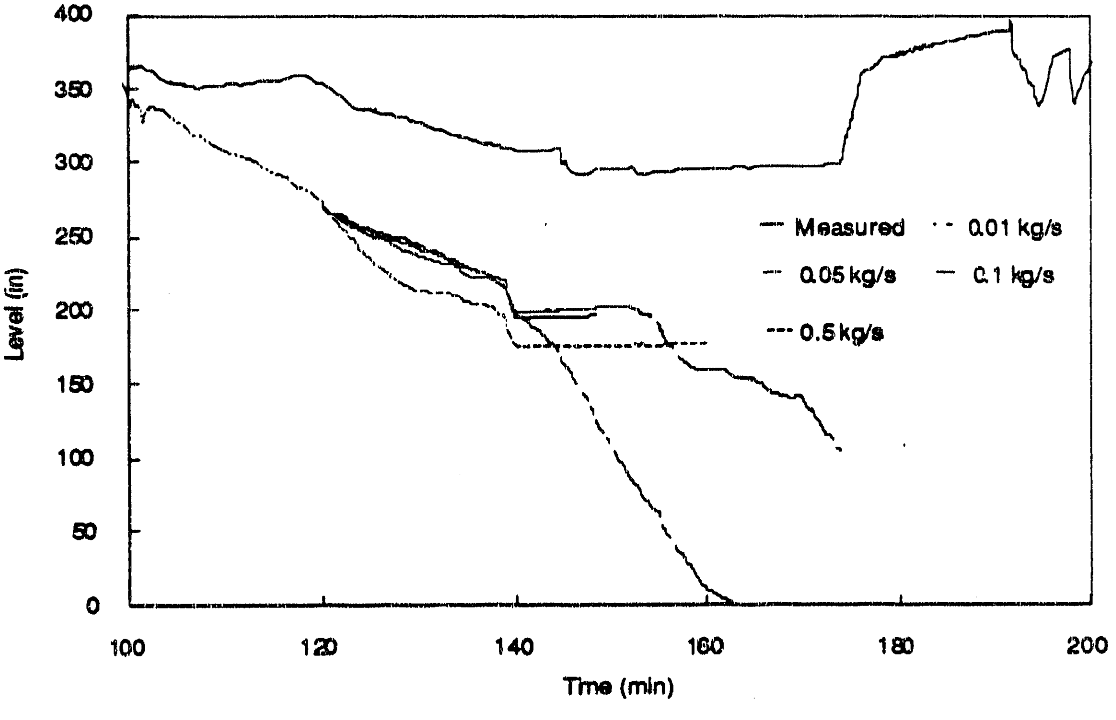

Figure 6. Calculated pressurizer level response to hydrogen injection.

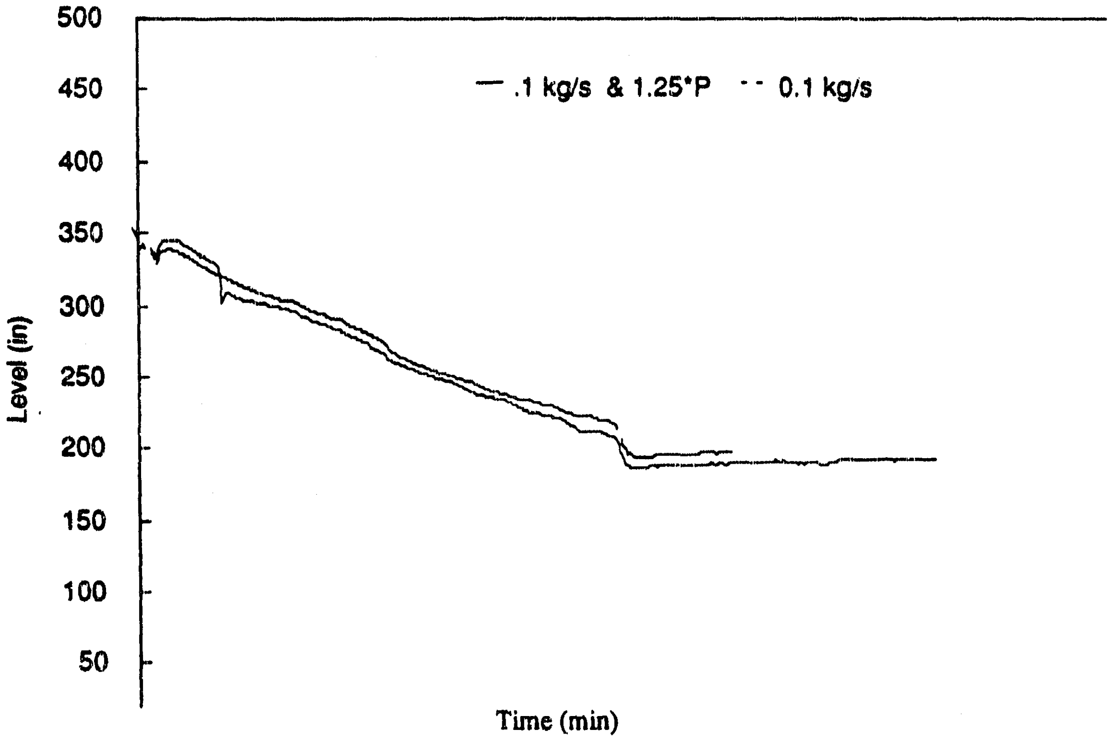

Figure 7. Calculated pressurizer response to simulated oxidation energy.

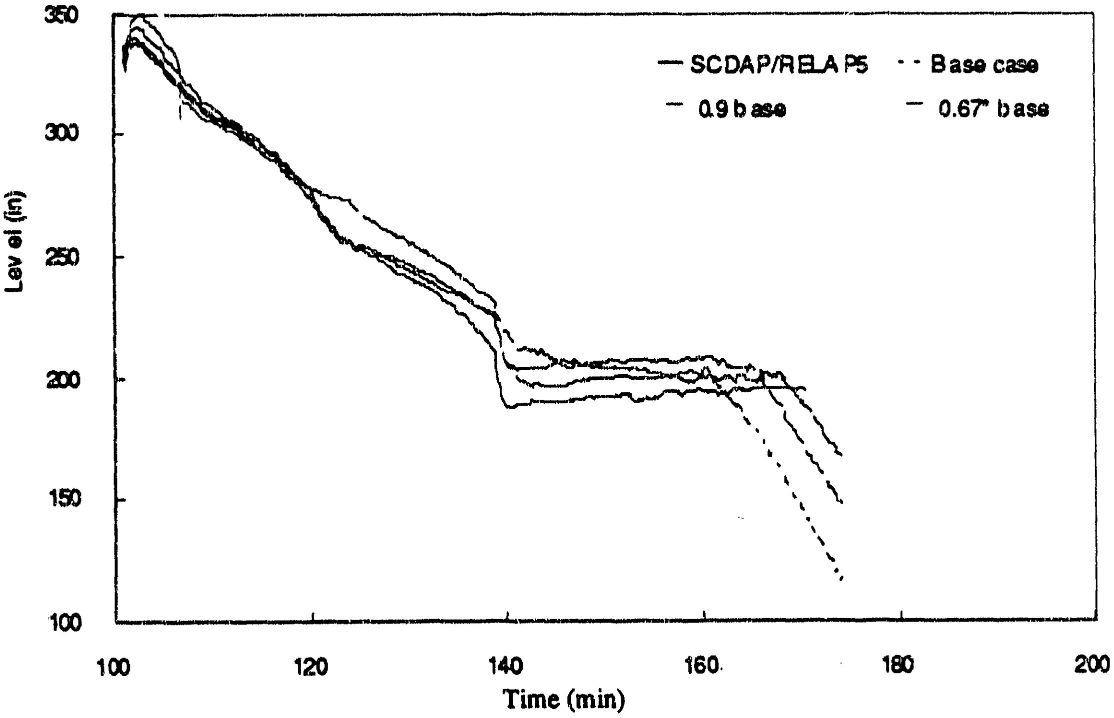

Figure 8. Calculated pressurizer level for steam generator heat transfer variations.

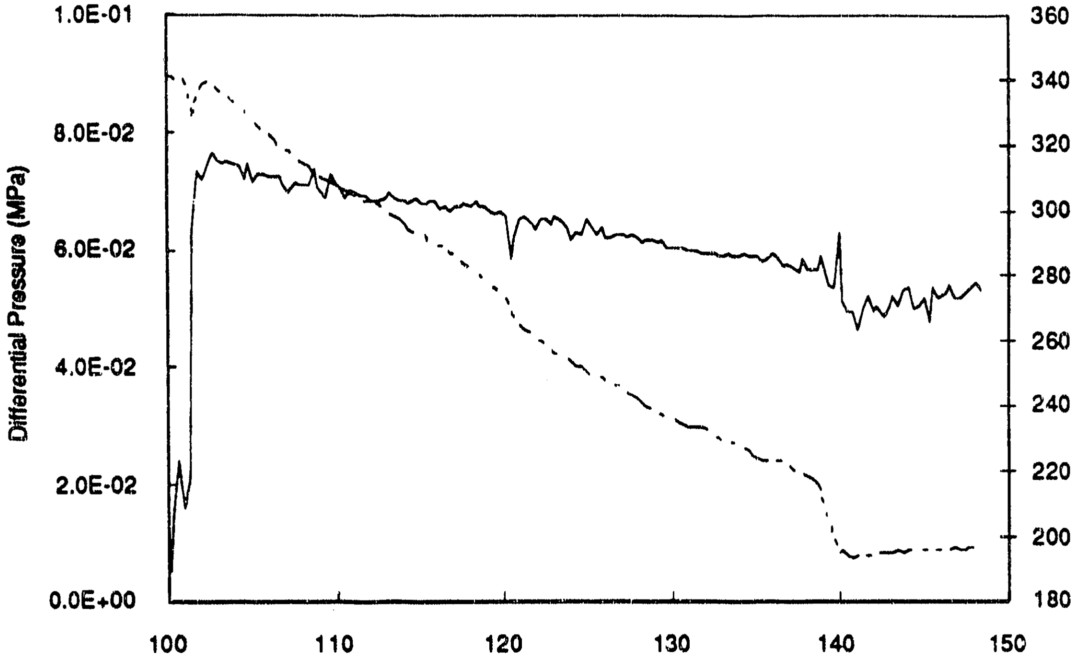

Figure 9. Hot leg-Pressurizer differential pressure vs. pressurizer level - 0.1 kg/s hydrogen injection.

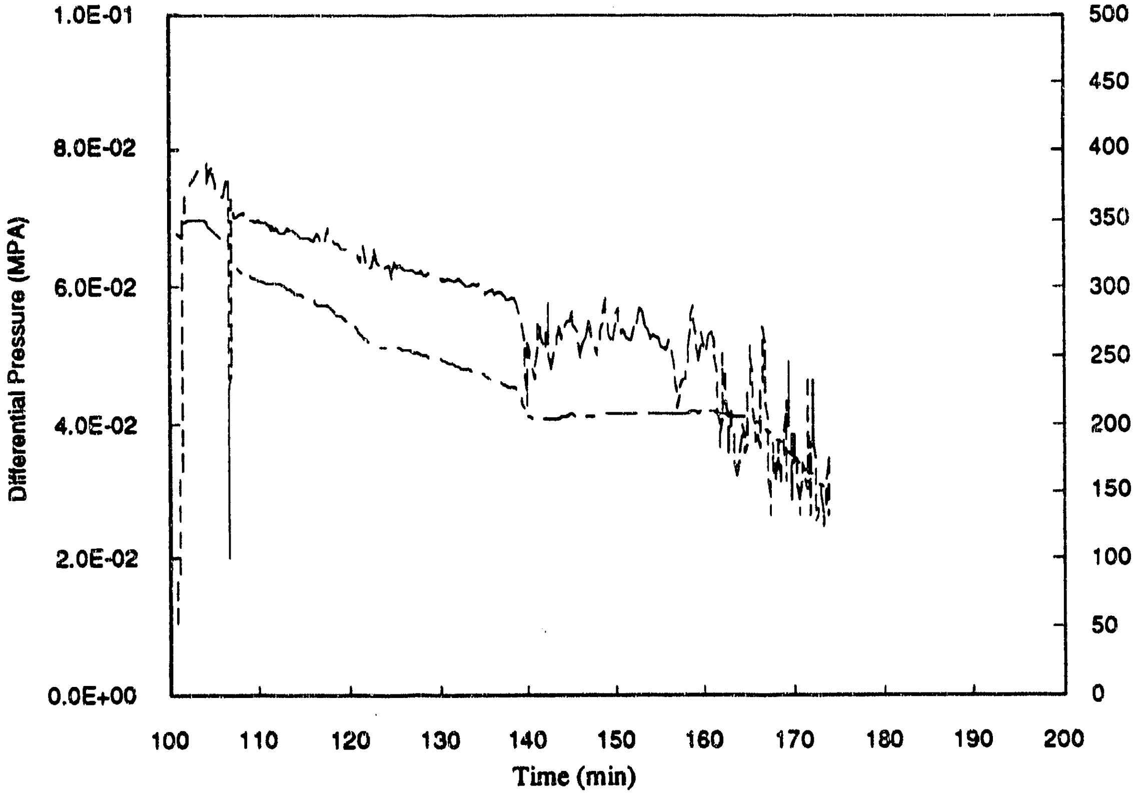

Figure 10. Hot leg-Pressurizer differential pressure vs. pressurizer level - no steam generators, 67 % heat transfer 100 min< time < 125 min.

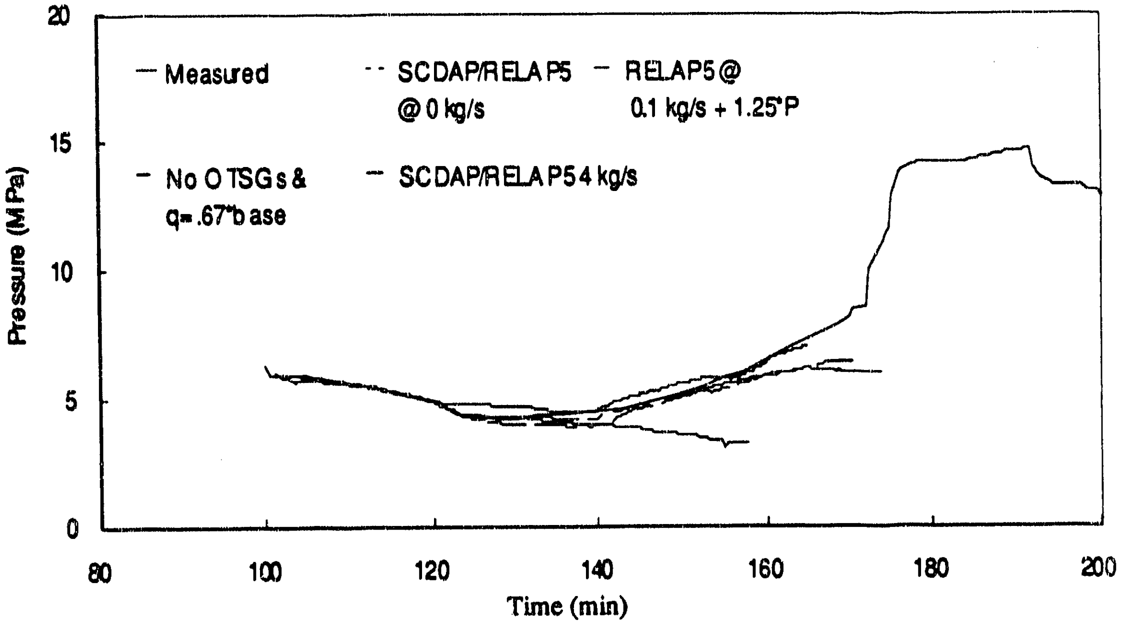

Figure 11. Calculated RCS pressure.

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