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PRELIMINARY APPRAISAL OF THE THREE MILE ISLAND ACCIDENT

The attached is for your information and use.

H. Etherington

Attachment:
Paper by H. Etherington re Preliminary
Appraisal of the TMI Accident

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Preliminary Appraisal of the Three Mile Island Accident

Introduction. The Committee has reviewed some of the data recorded during the Three Mile Island accident - the most important data are shown in attached Fig. 1. It seems clear that material balances and heat balances, in conjunction with recorded data not yet available to the Committee, will lead to an accurate history of the condition in the reactor, at least up to the time the reactor coolant pumps were tripped.

The following conclusions are supported by rough calculations which can be made available.

It is evident, as discussed later and as suggested by Curve F of Fig. 1, that, after the initial blowdown, the hot leg (at least) was at saturation temperature throughout the first 100 minutes. The void fraction is determined by the volumetric loss of primary system inventory. With no HPI or makeup pump, the loss of water and steam through the EMRV (electromatic relief valve) may be about 1 percent per minute of the system volume. This would also be the rate of void formation. The heat balance will reflect temperature changes necessary to generate the required steam. Thermal expansion and contraction is reflected in a change of void fraction of about 0.15 percent per ^{o}F (depending on amount of subcooling).

The pressurizer level. There has been speculation that the level instrumentation may be reading incorrectly. However, I believe the level indications are consistent with what is happening in the system, and it seems more reasonable to try to interpret what the instrument is seeing than to question its accuracy.

When voids are present in the system and the EMRV is open, all of the steam and water entering the pressurizer must be discharged through the valve. The pressurizer is thus filled to the top with a two phase mixture. In this conditior, what the level indicator is (probably) sensing is the weight of a two-phase mixture relative to the weight of water, e.g., a reading of 380 (Fig. 1 Curve A) indicates a void fraction in the pressurizer of 380/400, or 5 percent. To estimate the probably much greater void fraction in the reactor system, it would be necessary to know the electric heater input and the rate of rise of bubbles in the pressurizer.

The presumptions that arise from these considerations are that, with the EMRV open,

1. the level indication cannot exceed 400
2. the difference between 400 and the indicated levels is a valuable, but uncalibrated, measure of the void fraction in the system

On this basis, Curve A of Fig. 1 indicates the presence of voids throughout most of the accident, increasing from 40 to 100 min. The decreasing coolant flow shown in Curve C confirms this by the evident progressively increasing pump cavitation.

The period of forced circulation, 0-100 min.

1. The transients during the first 15 sec., following loss of main feedwater supply, follow a normal course, i.e., pressure rise, actuation of the EMRV, opening and closing of the safety valves, and other transients shown in Fig. 1. Failure of the EMRV to close is the governing factor in subsequent events. The effect of the delay in auxiliary feedwater flow was transient.

2. With the EMRV open, the system blows down to saturation pressure in a few minutes. Curve F of Fig. 1 shows that this occurs in 6 min., and calculation from the pressurizer steam inventory gives 7 min. During the first minute, the pressurizer level falls about 100 in. from the peak (Curve A). About 70 in. of this is accounted for by thermal contraction and about 30 in. by loss of pressurizer water by flashing during the blowdown.

The rise of 220 in. in the pressurizer level between 1 min. and 4 min. occurs before the temperature has dropped to saturation (Curve E). General voiding of the system is therefore precluded as the cause of the rise. No completely satisfactory explanation for this rise is offered. There has been a rise in the average system temperature (average of T_H and T_C, not adequately shown in Fig. F) of 2 to 3°F. This could account for 45 in. rise in pressurizer level. The strain relaxation of the steel boundary under a 400 psi pressure drop would account for only 1-1/2 in. Each of the three high pressure injection pumps would raise the level at the rate of 20 in./min. or 80 in. during the four minute period. The pumps could therefore easily account for the rise, but information on the pump history is confusing and the curve from which Curve A is drawn does not show discontinuities such as would be expected in starting and stopping pumps. Recent information suggests that one pump was running at capacity and one in a controlled mode. Voiding in the core is possible, but, with full flow circulation of subcooled water to remove decay heat, a large amount of core voiding does not appear likely. (I have made no calculations).

3. Thereafter, with the EMRV still open, the system will follow either of two courses:

> (a) It will remain at saturation, with the pressurizer level indication less than full, or
>
> (b) It will be pressurized to above saturation by "going solid" with the high pressure injection (HPI) pumps, in which case the pressurizer level will be over the 400 inch mark (with an adjustment for steam generated by the heaters).

It would be possible to operate in different modes at different times, but Fig. 1 shows that with the possible exception of the period 11-18 min. the system remained at saturation pressure throughout (Curves A and F). The following comments are therefore addressed to the saturated condition.

4. With the system at saturation, the fluid in the hot leg is two-phase, i.e., saturated water and steam. The steam voids increase progressively, with time, and cavitation of the reactor coolant pumps increased correspondingly (Curves A and C). It is concluded that misinterpretation of the level indication led to continued loss of inventory from the system.

5. The hot leg is at saturation temperature. After the first few minutes, the heat balance shows that, with all RC pumps running, the cold leg temperature is, as an upper limit, only about 1^{0}F subcooled. It is reasonably certain that, after the first few minutes, at least part of the core was functioning as a forced circulation BWR. It does not seem likely, however, that, at the low rate of decay, the core would have suffered damage during this period, even at the high probable void fraction. (I have made no calculations).

6. The system at no time appeared to suffer from serious loss of cooling capacity, except possibly during the period between 5 and 8 minutes. Curve F shows, with the steam generators dry, a rapid rise of temperature, and Curve B shows a corresponding rise of pressure. If the auxilliary feedwater supply had not been started, the pressure rise would presumably have continued until the safety valves opened.

7. Information available to us does not show when HPI ("makeup") pumps were started and stopped. It appears that the pumps can easily made up the EMRV loss and keep the system pressurized in a "solid" condition.

The period of "natural circulation."

1. Natural circulation in a PWR has been generally understood to mean circulation of liquid water. The conservative conditions for natural circulation in this mode are:

(a) The water must be subcooled

(b) The heat sink in the steam generators must be at a sufficient elevation above the core.

In Three Mile Island 2. the water was at saturation temperature, i.e., there was no subcooling. It should be noted that, when a reactor goes into natural circulation, T_H will increase and T_C will decrease, in order to give a temperature rise consistent with the reduced rate of circulation. It is not sufficient that the saturation pressure of the system be above the initial hot-leg temperature - it must be above the new hot-leg temperature..

The conclusion is that natural circulation did not, and probably could not, occur with the high void fraction, and the reactor sat in a pool-boiling state for 14 hours with no cooling except that supplied by the HPI pumps. I have not made any calculations to see whether this should have been adequate. The heat capacity should be adequate, but distribution and continuity of pumping would be controlling factors.

2. When the EMRV was isolated half an hour later, the pressure (Curve B) and saturation temperature (Curve F) rose gradually to a condition in which natural circulation might possibly have been induced.

3. Items that could have alerted the operator to the condition include:

(a) The wide spread between T_C and T_H. In forced circulation with all four pumps, this should be about 1^oF at 100 min. If the natural recirculation rate were 5 percent of this, the spread would be only 20^oF as compared with the wide spread indicated in Curve F.

(b) Availability of steam-table saturation temperature/pressure data proposed by Mr. Mathis. This would have shown, not only that natural circulation should not have been attempted, but also that T_H started to indicate superheat immediately, showing absence of water at the hot-leg temperature sensor.

(c) Immediate loss of steam generator pressure (Curve E).

4. These are other modes of natural circulation than the presumed design mode, but these should not be relied on without thorough analysis and testing.

Implications for Operating B&W Plants

The problem of the small break has been addressed by Mr. Michelson, Mr. Ebersole (Pebble Springs), and, more recently, Mr. Streeter of the NRC Staff in a memo that was still under review at the time of the accident. B&W has always responded with analyses that show the sytem to be capable of handling the accident. The analyses, however, assume no operator ill-advised intervention, and it is not clear that a complete spectrum of breaks has been covered.

The "NRC Status Report on Feedwater Transients in B&W Plants, April 25, 1979", summarizes the problem and status.

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FIG. 1 THREE MILE ISLAND ACCIDENT
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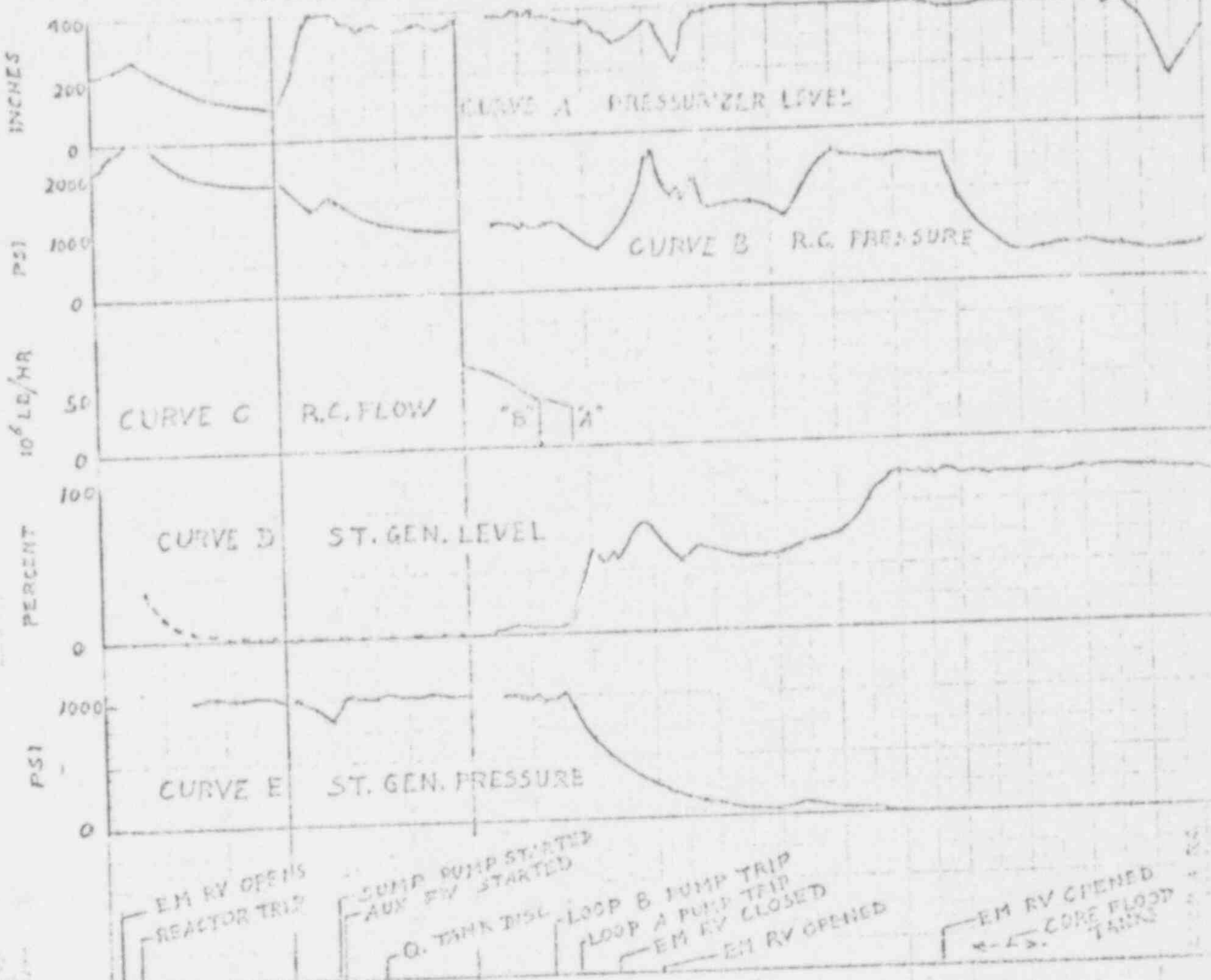

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