Review paper

The Accident at Three Mile Island

A. J. McEVILY*, Jr. and Iain Le MAY**

*Metallurgy Dept., University of Connecticut Storrs, CT 06269, USA

**Metallurgical Consulting Services Ltd.,

Saskatoon, Sask., Canada S7K 4E3

Abstract: This paper discusses the sequence of events involved in the accident which took place at the Three Mile Island Nuclear Plant in Pennsylvania, USA in 1979, the worst nuclear disaster in US history. The physical characteristics of the as-built reactor building are described, and the changes brought about by the accident on the plant structure are discussed. The current state of the plant and the materials of the reactor are also discussed.

Key words: Failure analysis, Three Mile Island, Nuclear power plants, Pressurized water reactor, Meltdown

1. INTRODUCTION

The Three Mile Island (TMI) nuclear power plant was built on a sandbar in the middle of Pennsylvania's Susquehanna River, just 10 miles downstream from the state capitol of Harrisburg. The principal structures on the island are two nuclear reactor buildings each containing two steam generators and four associated cooling towers. The reactor system is a Babcock and Wilcox 900 MW Pressurized Water Reactor (PWR). The system uses a once through steam generator and has an unusually small primary coolant system volume. The plant's state-of-the-art Unit-2 reactor had been generating electricity for nearly a year when a minor malfunction occurred on March 28, 1979 at 4:00 a.m. in the system which fed water to the steam generators at Unit-2. This malfunction led eventually to the most serious commercial nuclear accident in US history. Coincidentally, just twelve days earlier, a film called *The China Syndrome* had been released, providing a first look at a terrifying nuclear disaster. In the *China Syndrome* the core turns into a molten white-hot mass which penetrates the concrete base of the plant and reaches the ground water which is immediately below the foundation of the plant. Fracture of the earth then occurs instantly and geysers of radioactive steam are spouted into the air forming a radioactive cloud, similar to that created at Chernobyl, which leads to a widespread catastrophe.

2. THE TMI UNIT-2 REACTOR

The core region of the TMI Unit -2 reactor was contained within a pressure vessel made of low alloy steel of 4.3 m inside diameter. A concrete wall of 7 m inside diameter surrounded the pressure vessel. The purpose of the pressure vessel is to safely contain the core at the pressure and elevated temperature levels experienced in service. The concrete wall is a biological shield intended to absorb any fast neutrons and gamma rays emanating from the reactor. Dense materials are most effective for shielding, the three most commonly considered being water, concrete and lead. Concrete is an effective struc-

tural material, and provides a similar level of effectiveness in absorbing both gamma rays and neutron radiation. The pressure vessel and biological shield are housed within the containment building. This building provides additional biological shielding against radiation that might be released from the primary loop in the event of an internally initiated accident, and also provides protection against an internal event such as a gas explosion or an external event such as the crash of an aircraft.

Figure 1 is a schematic view of the Unit-2 reactor circuit and associated systems [1]. (The concrete shield surrounding the reactor has been omitted from this figure for clarity.) The unit had separate primary and secondary coolant systems [2]. The primary system was operated at a sufficiently high pressure, 15.2 MPa, to prevent reaching a bulk saturated water temperature and net steam formation during normal operation. At full power the pressurized water flows upward through the core at a rate of 1.74×10^4 kg/sec and reaches a temperature which ranges from 565 to 595 K. The water exits the core and reactor vessel and enters the tube side of the steam generators. At TMI-2 there are two steam generators with once-through flow on the primary side referred to as loops A and B. Each steam generator is fed by one hot leg and is emptied by two cold legs which return water to the reactor vessel under the driving head of the main coolant pumps. Steam to drive the turbine-generator units formed in the secondary system of the steam generators. Loop A contains the pressurizer and its relief valve that played a major part in the TMI-2 accident.

It is noted that the difference between a PWR and a boiling water reactor (BWR) is that, in the latter, the steam is produced directly from the water coolant in the reactor core and drives the turbine, before being condensed and returned to the reactor vessel. Thus there is no separation between the fluid in the reactor core passages and the fluid in the turbine, as there is in the case of a PWR. With the two separate loops of a PWR, any leakage of radioactive materials into the primary loop will be isolated from the turbine and will be confined to the containment area.

Fig.1. A schematic view of the Three Mile Island Unit-2 reactor facility [1].

3. DESCRIPTION OF THE CORE BEFORE THE ACCIDENT [3]

The fuel that was used was uranium dioxide (UO_2) powder that has been pressed, sintered and ground to form cylindrical pellets, 9.4 mm in diameter and 17.7 mm in length. To create a fuel rod, the fuel pellets were stacked inside zircalloy-4 cladding tubes, 3.9 m in length, 10.9 mm outside diameter, and 9.6 mm inside diameter. There were 208 fuel rods in each fuel assembly and the reactor contained 177 fuel assemblies. The assemblies contained three enrichment levels of U^{235} , 1.98%, 2.64% and 2.96 % by weight. The fuel assemblies were constructed with cladding covers to protect the fuel pellets from coolant corrosion. The overall diameter of the core was 3.27 m, and its active height was 3.66 m. Inconel 718 spacer grids were used to space the fuel rod assemblies at both the top and bottom of the core.

The control rods consisted of Ag-In-Cd in tubes of 304 stainless steel. Axial power adjusting rods of $\text{Al}_2\text{O}_3\text{-B}_4\text{C}$ clad with grade Cf-3M stainless steel were also used, together with two fuel assemblies which contained $\text{UO}_2\text{-Gd}_2\text{O}_3$ (gadolinia) test rods.

4. THE ACCIDENT

The accident sequence began at 4:00 a.m. on 28 March 1979 when the main feedwater pumps shut down (tripped), not an unexpected or unusual event [4]. This resulted in an automatic shutdown of the turbine and caused the emergency feedwater pumps to start up. Unknown to the operators, the block valves on both of the emergency feedwater lines had been left closed, so that no water was able to reach the steam generators. Since no feedwater was entering the generators, the reactor coolant temperature increased and caused an expansion which

raised the water level in the pressurizer. The pressure increase due to the compression of the steam in the top of the pressurizer, caused the pilot-operated relief valve (PORV) to open, and steam and water began flowing out of the reactor cooling circuit through a drain pipe to a drain tank on the floor of the containment building. The circuit pressure continued to rise and this caused an emergency shut-down, referred to as a SCRAM, of the reactor itself. However, since the core still had a residual heat generation of a few percent of full power caused by the decay of radioactive nuclides in the fuel, if the water level in the core drops enough to expose the upper portion of the core, the uncovered portion will heat up.

With the reactor shut down and the PORV open, the pressure in the primary circuit fell. The PORV was designed to close automatically when the pressure dropped to 15.2 MPa, and although the control panel indicated it had closed 13 seconds into the accident, it had in fact remained open so that coolant was draining off. This continued for two and a half hours before the circuit was sealed by the closure of a backup valve. Owing to the cooling of the core due to the shut-down and the loss of coolant through the PORV, as the water level in the pressurizer began to fall the operators turned on a pump to add water to the system. However, because the inflow was greater than the outflow through the PORV, the water level in the pressurizer began to rise. The rate of rise of the water level was increased when the steam generators, starved of feedwater because of the closed block valves, boiled dry and caused the reactor coolant to heat up and expand.

Two minutes into the accident the reactor pressure dropped to a value low enough to activate the high pressure injection system (HPIS), which is part of the emergency core cooling system (ECCS). The emergency cooling water was pumped into the reactor at a rate of

4000 l/min, and the water level in the pressurizer continued to rise. Because of this rise, the operators, who had been trained to avoid having the pressurizer filled with water, a condition which would make pressure control more difficult and might lead to plant damage, turned off one of the HPIS pumps and reduced the flow from the other. As a result, the water pressure in the reactor dropped to the saturation point, and steam bubbles began to form in the core. As a result the displaced water moved into the pressurizer, sending its level still higher. The operators, still responding to the perceived need to avoid a water-filled pressurizer, then began to drain off coolant. At this point, eight minutes into the accident, the closed block valves were opened restoring the feedwater supply to the generators.

One hour after the start of the accident the reactor coolant pumps began to vibrate badly, due to the mixing of cooling water with steam as a result of inadequate circulation in the core. The pumps were shut down in order to prevent damage to them or the coolant pipework. As a result all forced flow of water through the reactor was stopped. With the coolant continuing to escape through the open PORV, the top of the core became uncovered by water and heated up to a temperature where the steam reacted with the Zircaloy-4 cladding to produce significant quantities of hydrogen, some of which remained within the reactor vessel while the remainder escaped into the containment building.

Two hours and twenty-two minutes after the PORV had stuck open a backup valve was closed which terminated the release of coolant from the primary circuit. The damage to the core continued, however, since it was not until another hour had passed that the HPIS was restarted to attempt to make up for the loss of coolant that had occurred. At least two-thirds of the core had been uncovered and calculations indicated that parts of the core had reached temperatures in excess of 2200 °C, causing extensive fuel damage. At the four hour and twenty-six minute point the HPIS pumps were turned on again but it was not until two hours later that the core was completely covered. Attempts to establish natural circulation cooling failed because a large bubble of hydrogen gas in the top of the pressure vessel blocked the flow of water. At the seven hour and thirty-eight minute point the pressurizer valve was reopened and the rate of emergency coolant injection was cut sharply in an attempt to reduce pressure in the reactor. As a result a second loss of coolant occurred and the upper portion of the core was uncovered for an unknown period. After ten hours water was resupplied to the core, but it was not until sixteen hours after the start of the incident that the situation was considered to be under control. However it was not until almost one month later that Unit 2 was finally put into a natural circulation cooling mode to remove the last vestiges of decay heat from the damaged reactor core. Cold shutdown had at last been reached [5].

Meanwhile, about ten hours into the accident a hydrogen deflagration of undetermined extent occurred within the containment building. However, the pressure of 0.19 MPa generated by the explosion was insufficient

to cause any significant damage to the containment.

Then after three days into the accident the main concern centered on the possibility of an explosion of hydrogen inside the reactor itself which might rupture the upper head of the pressure vessel and lead to the release of radioactive gases. For such an explosion to occur, oxygen would have had to be present in sufficient quantity to form an explosive mixture. Initial calculations suggested that the rate of generation of oxygen by radiolysis would be great enough to create an explosive mixture, but this calculation was later found to be in error. In addition, the hydrogen overpressure (6.9 MPa) would have forced all of the oxygen produced to be recombined with hydrogen to form water. In any case, any danger of an explosion was removed when the hydrogen bubble broke up into smaller bubbles, which were readily eliminated.

5. FAULT TREE ANALYSIS [6]

It is instructive to analyze the accident causes and the allocation of responsibility using a fault tree approach, as has been done by Bradley[7].

Fault tree analysis was introduced in 1961 by the Bell Telephone Laboratories for performing safety evaluations for space-launch programs. The technique has since been used by reliability engineers as a general predictive tool for modeling system failures and system fault modes. This type of analysis is used to examine systems in order to determine component failure modes and other events (e.g., operator errors, maintenance problems, etc.) that can individually or in combination with other events cause system failure. The analysis is a top-down, deductive process whereby the analyst starts with the undesired state of the system as defined in the event tree and then proceeds to examine the system, component by component, for those combinations of component fault states that result in the undesired system state. The approach is opposite to the inductive method of first postulating a component fault state and then determining the result. Fault tree analysis produces a logic diagram depicting the component failure modes and other fault events that can, through AND and OR combinatorial logic, produce system failure. The fault tree is a binary model of the fault modes of a system and can be converted to a probabilistic model of the system - a model in which element failure probabilities are assigned to individual elements and combined to obtain system failure probabilities involving the use of Boolean algebra and probability theory (Boolean algebra is a logical combinatorial system that represents symbolically relationships as implied by the linguistic operators AND and OR between entities). The various symbols used in the construction of a fault tree are shown and defined in Fig. 2.

An example of fault tree construction will now be given. A loss of cooling accident in a PWR requires that the containment spray injection system (CSIS) operate following a reactor cooling system break. Failure of the CSIS is stated as "Insufficient Fluid Flow through CSIS Nozzles". This hardware definition is the undesired event

FAULT TREE SYMBOLISM

EVENT REPRESENTATIONS

The rectangle identifies an event that results from the combination of fault events through the input logic gate.

The circle describes a basic fault event that requires no further development. Frequency and mode of failure of items so identified are derived from empirical data.

The triangles are used as transfer symbols. A line from the apex of the triangle indicates a transfer in and a line from the side or bottom denotes a transfer out.

The diamond describes a fault event that is considered basic in a given fault tree. The possible causes of the event are not developed further because the event is of insufficient consequence or the necessary information is unavailable.

The circle within a diamond indicates a subtree exists, but that subtree was evaluated separately and the quantitative results inserted as though a component.

The house is used as a switch to include or eliminate parts of the fault tree as those parts may or may not apply to certain situations.

LOGIC OPERATIONS

AND gate describes the logical operation whereby the coexistence of all input events is required to produce the output event.

OR gate defines the situation whereby the output event will exist if one or more of the input events exists.

INHIBIT gates describe a causal relationship between one fault and another. The input event directly produces the output event if the indicated condition is satisfied. The term that permits the fault sequence to occur, and may be either normal to the system or result from failures.

Fig. 2. Standard fault tree logic and event symbolism [6].

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Fig. 3. PWR containment spray injection system TOP fault events [6].

Fig. 4. A simplified fault tree for the TMI-2 accident.

for which the fault tree is constructed, and it becomes the top event of the tree, see Fig. 3. There are two spray subsystems associated with the PWR, each of which is capable of supplying sufficient spray fluid to the containment. Both of these spray subsystems must fail in order to fail the system. Therefore the second level event "Insufficient

Fluid From Spray Subsystem A Header Nozzles" and "Insufficient Fluid From Spray Subsystem B Header Nozzles" are related to the top undesired event by an AND gate. Had both subsystems been needed to reduce containment pressure these second level events would have been related to the top event by an OR gate, indicating

that either event would fail the system. The next or third level events on the fault tree identifies the fault events that result in the second level events. That is "Insufficient Fluid Flow Through Spray Subsystem A Header Nozzles" can be caused by "Spray Header Ruptures", "Containment Pressure Sufficiently High to Reduce Spray Effectiveness", "Spray Subsystem A Header-Nozzles Plugged", or "Insufficient Fluid Flow to Header". The four faults identified are all of the immediate faults sufficient to fail the component header-nozzles. Similar faults can be identified for the subsystem B header nozzles. Additional faults needed to bring about these third level events can be developed in additional levels of the fault tree if desired. In Fig. 3, the event represented by the circle is a basic input event which is not developed further. The event represented by the diamond is a fault event, not necessarily a component failure, which is not expanded into all of its causes because there is insufficient available information to do so, or little additional information is expected to be learned by doing so. Rectangles represent faults for which the tree is developed further to identify more basic causes of failure. When completed the tree becomes a graphic representation of all component fault modes and other fault events that can fail the system.

Bradley [7] has used a modification of the fault tree approach in the failure analysis of disasters which have already occurred in order to work back to the fundamental human error which contributed to the failure. In this approach there are no OR gates in the fault trees, only AND gates. Bradley proposed the use of the following error codes in failure analysis.

ERROR CODES

Code	Error Type
B	Buying
C	Commissioning
D	Design
F	Failure of equipment, with no directly attributable human error
M	Management
O	Operating
P	Production
R	Repair

Code Suffixes

Suffix	Title	Explanation
1	Avoidable error	An error made by a person who should have known better, and in fact did know better
2	Unavoidable error	An error made by a person because the situation was outside of his knowledge or experience
A	Equipment failure	Equipment fails to operate when it should
b	Equipment failure	Equipment operates when it should not

A simplified fault tree for the TMI event is shown in Fig. 4, and the failure sequence is made clear by use of the fault tree technique. The fault tree developed by Bradley is more detailed, with the disaster code at TMI given by $F_a^3 F_b^2 M_1 M_2 O_1 O_2^3$, a complex combination of equipment failure, management errors and operating errors.

6. SUMMARY OF CORE DAMAGE [2]

Three years after the accident a robotic camera was lowered into the core for a first look at the full extent of the accident. It was found that five feet of the badly damaged core was gone; there had been a melt down. It was not the China Syndrome, but fifty percent of the core had been either destroyed or molten and something on the order of 20 tons of uranium had found its way, by flowing in a molten state, to the bottom head of the pressure vessel. It was observed that:

- (1) A large void was present in the upper core region with
 - a. A large and loose debris bed at the bottom of the void.
 - b. Remnants of fuel assemblies with partial and complete fuel rods radially surrounding the void.
 - c. Remnants of above-core components at the top of the void.
- (2) Just below the debris bed at about 40% of the core height was an apparently hard layer of unknown thickness and composition.
- (3) A large amount of fuel and non-fuel material has relocated to the bottom head of the reactor vessel, and much of the visible material does appear to have been molten.

7. EXAMINATION OF THE LOWER HEAD [8]

The lower head of the pressure vessel consisted of 136 mm thick A533B steel (0.25 C, 1.15-1.50 Mn, 0.035 max P, 0.040 max S, 0.15-0.30 Si, 0.40-0.70 Ni, 0.45-0.60 Mo; tensile strength 620-795 MPa, minimum yield strength 485 MPa; melting point 1515 °C) with a 5mm thick cladding of 308L stainless steel. Some 15-20 tons of molten core debris (2530 °C) had relocated there during the accident. In order to carry out the examination the lower head was cut into sections and the upper, most radioactive layer of the cladding was machined away. Nevertheless the steel was still slightly radioactive (9000 counts/min, Co^{60} , Cs^{137} , and Sb^{125}). Further material was removed by etching and abrasive grinding to bring the count to below 100, under which condition the material was considered to be safe for handling and metallographic examination.

In order to determine just how hot the lower head had been at the time of the accident, a samples of the same type of clad reactor-head steel were obtained. Specimens were cut from this sample and were heated at 800, 900, 950, 1000, 1050, and 1100 °C for 1, 10, 30, 1 and 100 minutes. These control specimens were polished, etched and compared with the TMI-2 boat samples. Slices from each boat specimen were hardness tested from the cladding to a depth of 40-50 mm beneath the interface. This

work revealed that the A533B plate material at locations near the center of the head had been reaustenitized and transformed to a higher-hardness structure upon cooling down after the incident. Metallographic examination of the control samples showed that the carbides present at the delta-ferrite/austenite interface in the cladding dissolve after 100 minutes at 1000°C or after 100 minutes at 1100 °C. In the base metal the prior austenite grain size began to coarsen at 1000 °C. Metallographic evidence of the steel of the lower head indicated that no melting or failure of the lower head occurred, and that the maximum temperature experienced at the inside surface had been 1075-1100 °C for thirty minutes. Further, the temperature 50 mm from inside surface had been lower by 100 +/- 50 °C.

The weld metal contained two features for quantification: the delta ferrite (required to prevent hot cracking) and the carbides in the delta ferrite. A modified Murakami's reagent (20g NaOH, 20g $K_3Fe(CN)_6$, 100 ml water) at 38 °C for 2-4 minutes was used to reveal the delta ferrite. Murakami's reagent (10g NaOH, 10g $K_3Fe(CN)_6$, 100 ml water) for 60-90 seconds at room temperature was used to reveal the carbides. An aqueous saturated picric acid solution containing a few drops of HCl per 100 ml and a wetting agent (sodium tridecylbenzene sulfonate) was used at room temperature for 5-10 minutes to reveal the prior austenite grain size of the A533B steel. At temperatures in the range of 1000 °C - 1100 °C the amount of delta ferrite in the control samples was found to decrease, as did the volume % of carbide within the delta ferrite. In addition the prior austenite grain size of the base material increased in this temperature range. By comparison with the TMI-2 samples it was estimated that the cladding may have reached a peak temperature of 1125-1150 °C for 30 minutes. In addition, in one specimen numerous fissures were observed in the A533B steel normal to the weld metal interface, and a few small voids were found in another specimen.

8. CURRENT STATUS

Today, the TMI-2 reactor is permanently shut down and defueled, with the reactor coolant system drained, the radioactive water decontaminated and evaporated, radioactive waste shipped off-site to an appropriate disposal site, reactor fuel and core debris shipped off-site to a Department of Energy facility, and the remainder of the site being monitored. The owner, General Public Utilities Nuclear Corporation, says it will keep the facility in long-term, monitored storage until the operating license for the TMI-1 plant expires in 2014, at which time both plants will be decommissioned.

In addition, analyses have been made of the events which would have occurred had the flow of coolant had not been restarted in the later stages of the incident. The meltdown might have stopped or it might have proceeded until the molten fuel had ruptured the bottom of the steel pressure vessel and dropped into the basement of the reactor building. A number of analysts agree that even with a core meltdown there was only a small probability that

the consequences of TMI would have been catastrophic to public health and safety. The most likely probability is that the reactor building would have survived (the building was designed to withstand a 333 km/h impact of a Boeing 727), and the vast majority of radioactive material released from the fuel would have been retained within the building, and not released to the surrounding environment, i.e., there would have been no China Syndrome [7]. There was a large financial loss however, with the cost of replacement of the TMI Unit-2 being valued at over \$1 billion [7].

9. CONCLUSIONS

(1) Despite serious damage to Unit-2, the release of radioactive material was relatively small and presented a negligible hazard to the public.

(2) The detailed causes of the accident continue to be debated to this day. However, based on a series of investigations, the main factors appear to have been a complex combination of equipment failure, management errors, and operating errors. It may also be that a part of the problem can be attributed to design errors because of lack of foresight into possible failure sequences. There is no doubt that the accident at Three Mile Island permanently changed both the nuclear industry and the Nuclear Regulatory Commission (NRC) in several ways, i.e.,

a. Public fear and distrust increased, and
b. NRC's regulations and oversight became broader and more robust, and management of the plants was scrutinized more carefully.

c. The problems identified from careful analysis of the events during the accident have led to permanent and sweeping changes in how NRC regulates its licensees, which, in turn, has strengthened public health and safety.

(3) The use of fault tree analysis in defining the responsibility for an accident has been demonstrated. This may be a valuable tool in a number of failure investigations.

(4) Based on the lessons from Three Mile Island, materials scientists and engineers have a much improved knowledge that even the most (apparently) unlikely failure scenarios must be addressed in the design of complex plants such as a nuclear reactor, and that operating personnel need to be better trained to deal with the unexpected. Better control procedures and systems must be in place to prevent the recurrence of such an accident.

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