Three Mile Island Accident

An Online Continuing Education Course for Engineers

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At 4 a.m. on Wednesday 28 March 1979, a main feed water pump at the Three Mile Island Unit 2 reactor plant shut down. Over the next several hours, a combination of mechanical failures and operator errors resulted in a partial reactor meltdown and release of radioactive material to the environment. Although the radiological exposure to the public was small compared to natural background radiation levels, a high degree of uncertainty at the onset of the accident, inadequate operator training, sub-standard communication, and a disorganized response by officials at the utility as well as state and federal agencies resulted in several poor decisions and lasting damage to the credibility of nuclear power in the United States.

Although the event initiated from a minor mechanical failure, the Three Mile Island accident was fundamentally driven by operator actions that aggravated and prolonged the disaster. Though some investigations may satisfy themselves that the operators were to blame, the fact that multiple well-trained and intelligent operators failed to recognize the underlying causes of the reactor accident for several hours and agreed to a poor course of action for a sustained period indicate that operator error was not the *root* cause of the accident. To be a root cause, operator error must be free of any causal factors in the underlying systems that prepare the operators for their task, direct them during operations, and aid them in making emergency decisions under the stress of a serious accident. As we will review here, such is not the case for the Three Mile Island event and is generally less likely than novice investigators may realize.

The lessons learned from this accident are applicable to virtually any high-tech engineered system where operators play a major role. Indeed, since few major engineered systems are designed to operate wholly independently of a trained group of specialists, technicians, and/or engineers, the lessons of the Three Mile Island accident will continue to resonate across engineering disciplines for many years.

The objectives of this learning module are

- Gain a general understanding of the pressurized water reactor system
- Review the details of the Three Mile Island accident
- Explore the major causes and legacy of the accident
- Consider the cross-functional learning points of the accident

Although not required, it is recommended that you take the introductory lesson “Learning from Engineering Disasters” before this course. If you have not previously taken "Learning from Engineering Disasters" or you need a refresher of the terms used, refer to the additional link in your Course Progress Page, Step 1 in your account.

Accident Analysis

Sidebar – PWR or BWR

All commercial nuclear reactors fall into one of two categories. The Three Mile Island design is a pressurized water reactor (PWR). The primary coolant system is maintained under high pressure to prevent the formation of steam. Heat is transferred to a different, non-radioactive coolant loop where steam is formed to turn the power generating turbines. This has the advantage of keeping the massive turbines free of any radioactive contamination. Boiling water reactors (BWR) combine these two loops into a single coolant system, reducing the amount of piping and eliminating the large and complex steam generator. This simplifies the overall system and reduces the potential for leaks. Steam formation in the reactor makes BWR nuclear dynamics very different from PWR's. The majority of US commercial nuclear reactors are PWR's.

We will examine the design and normal function of the Three Mile Island reactor plant, describe the failure at reactor unit #2, identify and categorize primary causes of failure, and establish the learning points broadly applicable across engineering disciplines. The Three Mile Island accident is relatively complex, but the opportunity for learning is also quite high due to the clear and broadly applicable lessons this failure teaches.

Understanding the Three Mile Island Nuclear Reactor System

The Three Mile Island reactor plant was a pressurized water fission reactor. In this type of power plant, nuclear fission is used as a heat source for a high-pressure water system circulating through the reactor core and primary coolant loop. Nuclear fission is a process whereby a large atomic nucleus is split by a neutron. Fission releases energy in the form of gamma rays, two or more smaller atomic nuclei, and one or more free neutrons that sustain the chain reaction. The gamma rays and neutrons interact with reactor plant materials and the primary coolant to heat the system.

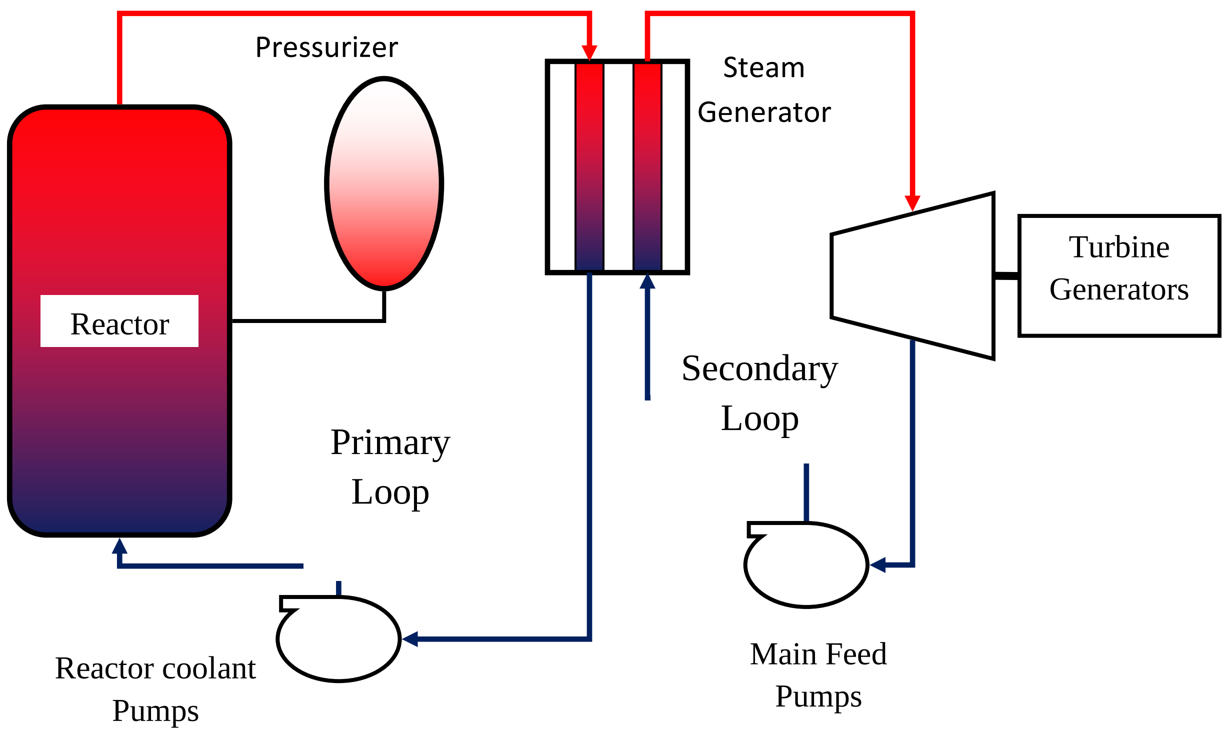
In the case of US commercial power generation, Uranium-235 is used as the nuclear fuel. Uranium-235 tends to produce two relatively large atoms during the fission process. These atoms are unstable and decay over a period of time (ranging from a few minutes to several thousand years) to more stable forms. This decay results in "decay heat" which is a tangible, significant amount of energy that must be removed from the reactor core following shutdown to prevent the primary system temperature from continuing to rise after the fission process is stopped.

The rate of the fission reaction is controlled by many factors but most especially the position of the control rods. These are made of a material that absorbs the free neutrons that sustain the fission chain reaction, preventing them from causing another fission event in the chain. The fission process may be stopped rapidly by inserting the control rods. An automatic rapid shutdown of the reactor is called a SCRAM. This protective action occurs when the control rods are driven into the core in a matter of a few seconds. It is important to note that because of the physics involved in nuclear decay a SCRAM only stops the fission chain reaction. It does nothing to prevent the release of decay heat. The fission byproducts created from Uranium-235 atoms that have already split continue to release energy independent of the fission reaction and regardless of the position of the control rods.

The primary coolant system is designed to maintain sub-cooled liquid water conditions under high pressure. This is accomplished by use of a heated water vessel called a pressurizer. The pressurizer is the only portion of the system intended to maintain saturated water conditions—the thermodynamic equilibrium allowing both liquid water and gas steam to exist at the same pressure and temperature. Liquid water in this vessel is heated to form some steam. Level detectors in the pressurizer indicate how much liquid water and steam is in the pressurizer to ensure that it does not fill with liquid water or steam alone. Pressure control is rapidly lost if either of these conditions arises. Since no bulk flow occurs between the pressurizer and the rest of the system, the relatively stagnant water in the pressurizer does not contribute to steam formation in the rest of the primary loop. The water in the rest of the system remains liquid at a lower temperature (heated from the fission reaction in the core but at temperatures below saturation) and high pressure (under the control of the saturated water system at a higher temperature in the pressurizer).

Fluid flow in the primary coolant system (which includes the reactor core) is driven by large reactor coolant pumps. The steam generator in the secondary system is fed by feed water pumps. All pumps are driven by electrical motors with redundant power systems. The highly pressurized primary coolant system gives up its heat to a secondary system operated at lower pressure. Steam is allowed to form only in the steam generator on the secondary loop side. This steam then turns a turbine, generating electricity for the commercial power grid. The lower pressure steam at the turbine outlet is then cooled and condensed into liquid water and pumped back to the steam generators by the feed water pumps.

Because the primary coolant system is pressurized to prevent steam formation, relief valves are installed near the top of the pressurizer to prevent over-pressurization of the primary coolant system. Pressurizer liquid water level and pressure throughout the primary coolant system will both rise as the liquid water is heated towards boiling. This phenomenon is due to the tendency of water to expand as it is heated. Changes in pressure occur slowly if steam is still present in the pressurizer to affect a “shock absorber” influence on the system. If only liquid water is present in the pressurizer, the system is designated as “solid” and responds very rapidly to small changes in temperature. If temperature rises or pressure falls in the primary coolant loop, saturation conditions may arise which would cause steam to form in the primary system outside of the pressurizer. This is highly undesirable because steam is a less effective medium for removing heat from the nuclear core than liquid water.



Summary of the Accident

The Three Mile Island accident actually began prior to the early morning hours of March 28th. Routine maintenance conducted on the secondary water system attempted to clear a blockage in the feed water system. Such blockages were common in the equipment that filtered minerals from the secondary loop feed water. This was done to prevent forming deposits in the steam generator left behind when water was boiled to feed the steam turbines. Such mineral deposits are generally undesirable because of their adverse effect on heat transfer in the steam generator and increased corrosion.

At 4:00 am on March 28th, this routine maintenance resulted in a main feed water pump tripping off. Investigative findings believe this was due to water introduced into control air lines during the previous maintenance activities due to the fact that a different method than normal was used to clean the feed system equipment this time. With feed water flow to the steam generator lost, the power turbine automatically shut down to prevent the steam generator from boiling dry.

Without a heat load from the steam turbine generator to draw steam from the steam generator, temperatures rose steadily in the steam generator, the primary coolant loop, and the reactor core. The pressurized water reactor was designed as a sealed system with only marginal ability to dampen pressure changes by compressing the steam in the pressurizer. With no excess volume to expand into, the heated water began to raise the pressure in the system. About 8 seconds from the onset of the accident, a pressure relief valve lifted to reduce system pressure and prevent fracture of the primary pressure boundary.

Soon after the pressure relief valve lifted, an automatic SCRAM action occurred, shutting down the reactor in a few seconds. Dozens of alarms immediately sounded in the control room. Because the reactor had been operating at 97% capacity, a large amount of decay heat continued to be generated by the core as fission products decayed into more stable isotopes. Estimates place the rate of heat generation at about 6% of total rated power for the reactor plant amounting to over 130 megawatts of thermal energy[1].

Three auxiliary feed water pumps automatically started to circulate feed water to the secondary side of the steam generator and continue to cool the primary loop. This should have allowed the primary system to continue to dump its decay heat to the secondary coolant loop and maintain a steady or lowering temperature in the reactor core. However, isolation valves on these auxiliary pumps were shut during maintenance and not re-opened, preventing water from reaching the steam generator.[2] With no flow of water to the secondary side of the steam generators, the residual secondary feed water rapidly boiled until the steam generators were empty. Within seconds after the beginning of the accident, the primary coolant loop was effectively isolated from its normal heat sink.

With the reactor shut down and the pressure relief valve open, steam vented from the pressurizer and reduced primary loop pressure. When pressure in the primary coolant system fell to a safe level, the pressure relief valve should have reseated. With primary loop pressure below the reset point, the solenoid that operated to open the valve de-energized as designed and extinguished an indication on the control room indicating panel. This control room indication for the solenoid was observed to go out as expected, but the actual status of the valve was still open. A material failure resulted in the valve sticking open although the control solenoid was now de-energized. The operators incorrectly took the indication of the solenoid being de-energized as proof that the valve was shut while alternate available indications clearly showed that there was still a steady flow of water through the open relief valve. For the next two hours and twenty-two minutes, this failure would result in a steady leak of primary coolant from the reactor core resulting in a loss of roughly one-third of primary loop coolant[3].

Due to this leak, pressure in the primary coolant system continued to drop while decay heat continued to raise temperature. The emergency cooling system for the reactor plant (high pressure injection or HPI) activated automatically at 1000 gallons per minute to prevent thermal damage to the reactor. This rapid injection of cooling water from the emergency system led to a rise in the level of water in the pressurizer but did not force pressure to rise due to the ongoing leak. Observing the rising water level in the pressurizer, the operators believed the primary

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[1] The 6% of decay heat produced after SCRAM is cited in the Report of the President's Commission. The design thermal output of the reactor is reported as 2272 MW thermal in NUREG-CR-6197 part 2.

[2] The report by the President's Commission and other sources concede that these valves may have been inadvertently shut during the confusion of the opening minutes of the accident. However, due to corroborating reports of deficient maintenance practices in place at the Three Mile Island reactor plant, the author believes other investigative findings that these valves were never re-opened following maintenance performed in the days prior to the accident.

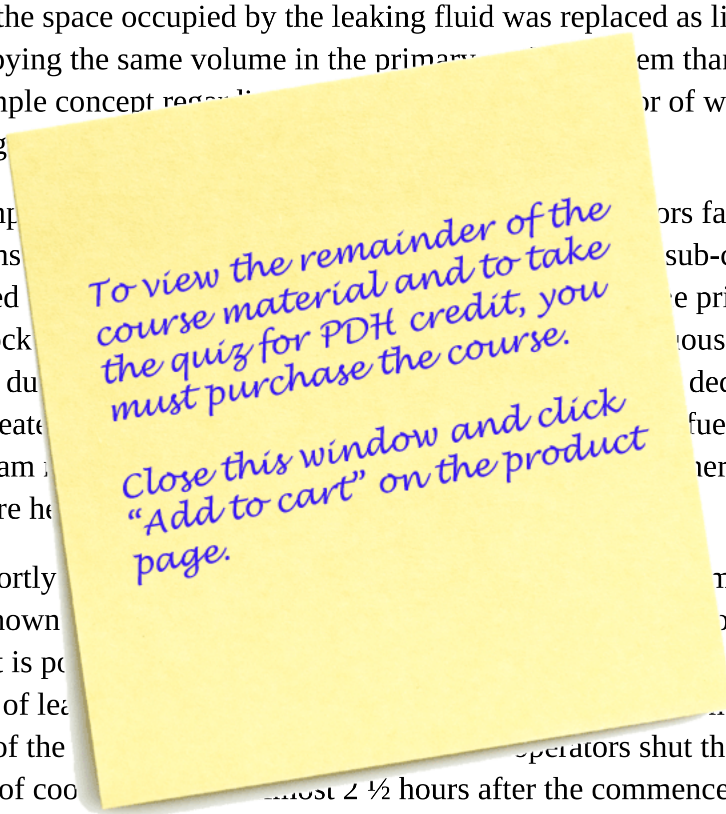
[3] Report of the President's Commission On the Accident at Three Mile Island, 30 October 1979, pg 91.

coolant loop now had sufficient water in it. Knowing a full pressurizer would result in a loss of pressure control, the operators decided to secure one HPI pump and throttle flow on the other to less than 100 gpm. This pump would be shutdown as well within the next two hours, completely eliminating all sources of water to replace the leaking coolant.

Temperature in the primary coolant system soon stabilized with pressure continuing to fall. Alarms sounded due to rising water in the containment building sump and temperatures on the pressure relief valve outlet was observed to be significantly above normal. Though action was taken to address these *symptoms*, the operators failed to recognize the unifying *cause* of these problems—that a leak of some sort had developed in the reactor coolant system.

Approximately one hour into the accident, the reactor coolant pumps began vibrating significantly due to pumping a liquid-steam mixture instead of sub-cooled liquid water. Fearing damage to the pumps, the operators performed as trained, securing two of the pumps immediately and the other two thirty-seven minutes later. Operators believed erroneously that natural circulation due to thermal gradients in the system would continue to keep water flowing through the primary coolant loop and prevent damage to the core.

What was happening, in fact, was a steady rise in the core temperature resulting in the formation of a steam pocket in the reactor vessel. This steam pocket was responsible for the misunderstood behavior of the pressurizer. Despite a steady loss of fluid from the pressurizer through the stuck open relief valve, the space occupied by the leaking fluid was replaced as liquid water vaporized into steam—occupying the same volume in the primary ... em than the liquid water it replaced. This simple concept re... ...r of water contributed to the confusion masking ...



Despite taking temp... ...ors failed to recognize that saturated conditions ... sub-cooled water system steadily approached ... e primary coolant system. Once the steam pock... ...ous and any possibility of natural circulation du... decay heat generated in the core steadily heate... fuel itself was blanketed by steam. The steam ... ...erate zirconium oxide, hydrogen, and more he...

At shift change shortly ... ...mencement of the accident—an unknown ... ...ocking valve to isolate a potential leak. It is po... ...rator, but it may also have been a result of lea... ...pted to assist in regaining control of the ... ...perators shut the blocking valve to terminate the loss of coo... ...most 2 ½ hours after the commencement of the accident.