A REPORT

TO THE MET-ED COMMUNITY

Metropolitan Edison Company
Reading, Pennsylvania
May 30, 1979

Report Number Two

Contents

Dear Neighbor,

Here is additional information relative to the accident at Three Mile Island. It is in the form of a report on the accident as presented May 9, 1979, at the GPU Annual Shareholders' meeting.

We sincerely hope that this information will help build greater public understanding. We are fully aware that everyone close to these events has concerns. Met-Ed wants to be responsive and we urge you to write to let us know of your special interests. We plan to continue this series of reports to the community and I can assure you that Met-Ed will make every effort to address your concerns in future communications.

Sincerely,

Walter M. Creitz
President

THE TMI-2 ACCIDENT

Herman M. Dieckamp, President
General Public Utilities Corporation

as presented to

GPU Annual Stockholders Meeting
May 9, 1979, Johnstown, Pa.

I would like to provide a brief description of the major pieces of equipment that constitute a nuclear plant like Three Mile Island, so that as you read about the accident in the press and other media, you will be better able to follow the discussion and understand what is being said.

I will then provide a brief description of the components of the system and the manner in which the response of TMI's operators contributed to the magnitude of the accident. I will finish off with a brief description of the status of the plant and the outlook for the future.

I certainly don't have any aspirations to convert all of us to senior nuclear engineers, but I do think these are things that can be described and can be understood. Let me see if I can lead us through a nuclear unit.

If you look to the left side of the visual aid (enclosed), you will see what we refer to as the primary coolant circuit of the plant. That portion is all contained within the reactor containment building, represented by the dashed line with the label up towards the top with the caption that says, "Containment." When you look at pictures of nuclear plants you will see generally a large cylindrical concrete structure -- that is the containment building -)f the plant.

Now let's look at some of the components within that primary cooling circuit. First let's start with the reactor core. It is there within the reactor vessel. The reactor core is a region of nuclear fuel assemblies. The core is, roughly, 10 feet in diameter and about 12-13 feet in height. The reactor itself is a heavy-walled pressure vessel that is of the order of 60 feet high and about 15 feet in diameter.

The nuclear reaction produces heat. In that sense, it is no different than a fossil fired power plant. We start with a form of energy, that is, heat which we in turn want to convert to electricity. That heat is transported by means of circulating high pressure and high temperature water.

The primary system runs at about 2,000 pounds per square inch and at about 600°F. The primary coolant pump causes that water to circulate through the system. The purpose of this circulation is to transport the heat from the reactor to the steam generator.

In the steam generator, water from a separate loop (that one which starts out in the sector outside the containment building, the right hand portion of the schematic drawing) is caused to boil and absorb the heat from the reactor in the form of high-pressure and high-temperature steam. That steam in turn then is expanded through a turbine and then is discharged fran the turbine to a condenser. The steam is cooled, "condensed" and changed back to water. It then is returned by means of feed water pumps to feed water heaters and then back to the steam generator. In many ways, this secondary portion outside the containment of the plant is not at all unlike a fossil-fired power plant.

One other component that I would like to point out at this time is back in the primary loop and is the one labeled "Pressurizer." Its purpose is two-fold: 1) it is the device by which we maintain the high pressure in that primary circuit and attempt to maintain that cooling water at all times in a liquid or non-boiling state; 2) that pressurizer is used to absorb changes in volume as the primary system heats up and cools down.

I think this largely describes the functioning parts of the system that you have seen referred to in various press accounts of the accident and, I would hope, gives you sane feel for how the power plant functions.

Again, briefly, the heat in the reactor is transported by flowing water to the steam generator. The secondary loop of water is converted to steam in the steam generator, expands through a turbine, the turbine turns a generator and makes electricity. The discharged steam from the turbine is condensed and returned to the steam generator. The heat energy which is not converted to electricity is rejected to the environment in a third loop through the massive cooling towers that you see dominating photos of the TMI station.

Now, let us turn our attention to the things that did not function as intended and, thus, contributed to the magnitude of the accident at Three Mile Island.

First of all, the accident began with a failure in the secondary, non-nuclear portion of the plant. Specifically, the main feed water pumps were turned off by some mechanical or electrical failure in their control circuitry. That, in turn, led to a reduction in the heat removal capabilities of the steam generator and, as a result, not only did the turbine trip (by that we mean it was shut off), but also the reactor tripped or was scrammed (by that we also mean it was shut off). These two events occurred very rapidly and exactly as expected.

At this point everything was in accordance with normal design. Immediately, however, the pressure in the primary system (the nuclear portion) began to increase. In order to prevent that pressure fran becoming excessive, a valve located at the top of the pressurizer opened up. That valve (you will sometimes find it referred to as an electromatic valve) should have reclosed when the pressure decreased by about 100 pounds per square inch. However, that valve failed to reclose.

The signals available to the operator, both in terms of an indicator of the canmand to close and in terms of temperatures in the region of that valve did not indicate to the operator that the valve continued to be open. However, the fact that that valve was stuck open caused the system pressure to continue to decrease.

Next, let's turn to the emergency feed pumps. The emergency feed pumps are backup duplicate safety devices in the event the regular feedwater pumps fail. They are subjected to routine surveillance tests in order to determine that they are functionable and are available to support the plant in case of need. The last time that system was tested was 42 hours prior to the accident.

In order to test that emergency feed system it is necessary to close a valve and isolate it so that it cannot open. The test program requires that that valve be reopened into the safe condition at the end of the test program. Through some administrative or human failing, that valve apparently was not restored to the open position at the end of the test. It was discovered as being closed about 8 minutes after the start of the accident. The operators then opened that valve and that system functioned as intended.

During this period, the system pressure continued to diminish due to leakage from the open pressurizer valve.

In response to a reactor and turbine scram, as was experienced here, the operator knows that water levels in the primary loop normally begin to decrease. He has available to him a gauge which measures the water level in the pressurizer. Under normal circumstances, that water level is his prime indicator to tell him that the primary system is full of water and thus capable of reliable heat transfer or heat removal from the core.

As the system pressure continued to decrease due to the stuck valve, voids began to form in portions of the system other than the pressurizer. Thus the liquid in the primary system redistributed itself and the pressurizer became full of water, but there were voids in other parts of the system. The level indicator in the pressurizer suggested that the system was full of water and caused the operator to stop adding water to the system. He was unaware that, because of the stuck valve, the indicator can, under some circumstances, became ambiguous.

The net result of this continuing reduction in pressure and the halt of additions of water to the system (halted because the information available to the operator suggested to him that the plant was adequatel-, full) caused the development of steam voids in the primary loop where there should be only water. This reduced the efficiency of heat removal from the core.

About 100 minutes after the start of the accident, the operator noted that the main pumps were getting to a region of operating conditions that were beyond their defined limits. As a result, he turned off the four circulating pumps. This had the effect of further diminishing the ability of the system to remove heat from the core.

About 100 minutes to 200 minutes after the accident, the removal of the residual heat being produced in the reactor core was inadequate. Because of this, the fuel materials overheated to the point that **sane** of the zirconium cladding (that contains the nuclear fuel pellets) reacted with water and generated hydrogen. The hydrogen, in turn, was released to the reactor containment building. Sane hydrogen remained within the primary coolant system and resulted in the hydrogen bubble we heard so much about.

The real damage to the reactor occurred in this time period of 100 to 200 minutes after the 4:00 a.m. start of the accident. The operators required until about 8:00 in the evening to return the system to a near normal operating condition, with the primary system full of water (except for the hydrogen bubble) and with the pumps operating so as to have firm reliable heat removal fran the reactor core.

So, without going further into detail, let me say in summary, that the accident was a result of a complex combination or interaction between equipment failures, procedural failures, operator misjudgments, ambiguous instrumentation and a number of factors which all, when contributing together, led to this problem. In order to fully understand the role of the operators, as contrasted with the role of the equipnent failures, one has to look back at our prior conceptions of reactor accidents and the degree to which they formed the foundation for training and the degree to which the operator's prior experience preconditioned his responses. I think it is clear to us, and we are confident that the many subsequent investigations will confirm, that the accident was not a simple case of an operator who made a mistake but, rather, that the accident was a result of a complex interaction of an unanticipated combination of factors.

Now let me go ahead to say where the plant is today. The plant has been put into essentially what has been popularly referred to as the cold shut-down condition. The residual heat from the reactor is being removed by means of natural circulation of the water in the primary loop. The heat is being removed by sending water through the steam generator so that it produces steam which, in turn, goes to the condenser. In the condenser the steam is cooled, the heat rejected through the cooling towers. The maximum temperature inside the reactor core is reported by the NRC to be about 310 degrees fahrenheit. The average temperature of the water that is circulating to cool the reactor is about 170 to 180 degrees fahrenheit. This is below the boiling point of water, even at atmospheric pressure.

Backing up, let me emphasize that in the immediate time period after the accident, our attentions were directed to four high priority activities. One, to maintain the crippled or damaged reactor in a continuingly safe operating mode so that the situation would not further degenerate. Second, to do everything humanly possible to minimize any releases of radioactivity to the environment, and thus any hazard to the local populace. There were, indeed, some releases of radioactivity. They were, to a large extent, the result of continuing safety operations that had to be done within the plant. A number of modifications were made to the plant in order to provide additional filtration devices to capture any radioactivity and minimize releases to the public. The third goal was to move safely to cold shut-down. And we have recently completed our fourth priority activity, to put in place a number of auxiliary systems to reinforce the ability of the plant to remain in this safe cold shut-down condition.

As we look forward, we think the plant will be out of service for approximately three years. During this time, we will need to remove any radioactive atmosphere from the primary containment. We will have to reenter that containment and begin to decontaminate any radioactive materials that spilled out through the stuck open relief valve. We will then have to gain entry to the reactor vessel, ascertain the exact degree of mechanical damage to the fuel material, remove that fuel material and then clean up the primary loop. These activities then are the necessary precursors to returning the plant to service.

I would hope that this gives you a brief run down of the plant, the accident and the status that we are now in.

Thank you.

THREE MILE ISLAND UNIT TWO SCHEMATIC