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THE TMI ACCIDENT, WHAT REALLY CAUSED THE CRISIS?

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ABSTRACT

Although the Three Mile Island accident was the most serious which has yet occurred in the U. S. commercial nuclear power industry, the real radiological health effects were minimal. However, a combination of misinformation and sensationalistic media coverage turned this situation into a crisis of major proportions which continued unnecessarily for many days.

This paper summarizes the various aspects of the accident. This includes a discussion of (a) the radiological consequences of the accident, (2) the events which led to the uncovering of the core and subsequent recovery, and (3) the circumstances and misinformation which caused the unnecessary NRC evacuation recommendations and the unnecessary concerns about a hydrogen explosion in the reactor vessel.

Most of the discussion, especially concerning the misconceived evacuation and hydrogen explosion potential is from the vantage point of someone who was uniquely and intimately involved in most aspects of the offsite response to the accident. It is, therefore, a corcise chronology of the accident and the factor which turned it into a crisis.

NOTE: This paper was prepared during the later part of September, 1979, about one month prior to the issuance of the President's Commission Report on the TMI Accident. It is a composite of various talks given by the author to both technical and non-technical groups on the TMI Accident. The paper was written in an effort to bette give factual information on this matter to a largely misinformed public.

The accident at the Three Mile Island Nuclear Station, Unit 2, which occurred on March 28, 1979, has received the most notoriety and the largest amount of media coverage of any event in recent years. Although this was to date the most serious accident which has occurred in the commercial nuclear power industry, the real health hazards were minimal. In addition, the chances of a catastrophic core meltdown or other eventualities which might have increased the severity of the health risks were much less than that which has been painted by the sensationalistic media coverage which this event precipitated. This is the major reason that I feel it necessary to relate my experiences and interpretation of what really happened those first few days because the reported version caused what I feel to be a grave injustice to the people of Central Pennsylvania.

I am a nuclear engineer employed by the Pennsylvania Bureau of Radiation Protection. This agency according to the State's Emergency Plan had the prime responsibility for recommending protective action in the event of a radiological accident, or so we thought. I am also the only nuclear engineer employed by the Commonwealth, so I therefore had a very unique perspective with which to view the events as they occurred.

For the first three days, I was in a position to communicate directly with the Med Ed plant personnel and the NRC I & E inspectors who arrived on site within the first few hours. During this time I also participated in most of the meetings which occurred in the Governor's Office and the other decision making processes which were occurring at the state level. After Friday when Harold Denton arrived on site, I was assigned the task of being the onsite liaison with the NRC and had the responsibility of informing the Governor's Office and other state agencies of actions being taken or contemplated which could have offiste significance. In this position I was privy to all the information that was available to NRC. I was also able to learn firsthand from the NRC personnel who were actually involved, the unfortunate events which precipitated the unnecessary concerns about evacuation and the potential for a hydrogen explosion in the reactor vessel.

With these thoughts in mind, the following is a brief summary of (1) the radiological consequences of the accident, (2) the design, mechanical and operational errors that caused the accident, and (3) my personal involvement and the actual circumstances which caused the evacuation and hydrogen explosion concerns.

It is probably appropriate to begin the discussion of the radiological consequences of the TMI accident by noting the monitoring devices which were present in the environment around the plant before the accident and those which were added as the accident progressed. Figure 1 is a map of the area within 20 miles of the site which shows the continuous air sampling devices and the milk sampling locations which were established prior to the accident. Figure 2 shows locations of the thermoluminescent dosimeters (TLD's).[1] The Met Ed and

Commonwealth of Pennsylvania TLD's were in place prior to the accident,[2] while the NRC and EPA TLD's were placed after March 30, which was also after most of the releases had occurred.

In addition to these essentially permanent monitoring locations, starting early Wednesday, mobile monitoring teams from Met Ed and the Commonwealth of Pennsylvania were performing beta gamma surveys and taking portable air samples almost continuously at various locations around the plant. These teams were supplemented later in the day on Wednesday by NRC and DOE teams which performed similar surveys up and down the east and west shore of the river on a continuous basis. As an additional supplement, helicopter teams from Met Ed and DOE periodically, and when specifically requested, performed airborne surveys primarily to quantify and define the plume of radioactive noble gases which were being released in varying amounts almost continuously over a long period of time. Numerous samples at different locations of other media such as river water, soil and vegetation were also analyzed to assure that nothing other than airborne noble gases were being released in significant quantities.

Based on the monitoring data, the NRC has estimated that about 13 million curies of radioactive noble gases and about 14 curies of radioactive iodine were released as a result of the accident.[3] This amount is many times that allowable by the NRC in the unit's technical specifications. Also based on the monitoring data it has been estimated that the maximum cumulative dose received by any member of the public due to noble gas emissions was about 83 millirem. This is conservative because it assumes that the individual remained at the same location, out of doors, with no clothing for a period of about one week.[4] The corresponding maximum possible dose to the thyroid due to inhaling radioiodine or drinking the milk with the highest found contamination[5] is estimated to be less than 5 millirem to a child's thyroid.

A further evaluation of the maximum cumulative doses received by the population, based primarily on the TLD data for the first week, is shown in Figure 3. These isodose curves show that out beyond about 10 miles the maximum cumulative dose was less than 1 mrem during the first week.[6] This compares with a natural background radiation dose in this general area of about 2 millirem over this same period.

To give a perspective on the relative magnitude of the releases over the first few days, Figure 4 is a plot of maximum individual and population dose verses time. It becomes evident from this figure that by Friday noon when the pregnant women and children advisory was given, about 90% of the individual dose would have already been received. Therefore, the evacuation in addition to being unnecessary was also not very effective.

The most comprehensive study[7] to date, of the health effects of the accident, was performed by a task force of radiological health professionals from the NRC, EPA and HEW. The balance of the information in this section is taken from that report, a summary of which is included as Appendix A.

The major conclusions of that report are the following:

(1) The maximum cumulative dose that an individual located offsite might have received is less than 100 millirem.

(2) The estimate of the collective dose to the population of about 2 million within 50 miles of the site range from 1500 to 5300 person-rem, with the most likely estimate being about 3300 person-rem.[8]

(3) To provide a perspective on the dose received as a result of accident, Table 1 shows a comparison with some natural background radiation exposure.

(4) In order to estimate the number of expected health effects as a result of the accident, Table 2 gives the fatal cancer and genetic effects risk factors.[9]

(5) Finally, the actual expected health effects over the lifetime of total population within 50 miles is given in Table 3. This is compared with the total expected fatal cancers in that population along with those expected from natural background radiation.

It can therefore be concluded that the radiological consequences of the accident were indeed minimal. However, the psychological stresses and anxieties which were created mainly because of the misinformation and the sensationalistic media coverage could have produced some very adverse effects on the population, many of which will be very difficult if not impossible to quantify.

The design philosophy for a nuclear power plant requires the use of several independent barriers, all of which must be violated to allow the release of significant quantities of fission products.[10] In the case of the TMI accident most of these barriers were at least partially breached for limited periods of times for various reasons as follows:

(1) The first and probably most important barrier is the fuel rods as shown in Figure 5. This barrier is a combination of the ceramic uranium oxide fuel pellets along with the zirconium alloy cladding in which they are encased. In order for a large fraction of the fission products which become trapped in the fuel pellet matrix to escape, the fuel pellet must melt and the cladding must be breached. In addition, during operation a small fraction of the more volatile fission products such as noble gases and iodine migrate out of the fuel pellet and become trapped in the gaps at the end and between the pellets. Therefore if only the cladding were to fail this "gap activity" would be released. This was primarily what happened during the accident. Due mainly to an operational error and a misinterpretation of the instrument indications, the operators did not maintain sufficient inventory in the reactor coolant system and the core eventually became uncovered. This caused some of the fuel rods to increase in temperature to a point where a zirconium metal-water reaction occurred. This reaction eventually caused a breach of the cladding and generated a significant amount of hydrogen.

(2) The second barrier to the release of substantial amounts of fission products, assuming the fuel rod barrier is breached, is

the reactor coolant system. An elevation and plan view of this system is shown in Figure 6. This barrier was breached during the first two hours of the accident due to a mechanical failure of the power operated relief valve. This valve, which is located on the pressurizer, failed to reclose after it had opened on increasing pressure in the system following the initial turbine trip and loss of feedwater transient.

(3) The final barrier to the release of radioactive material in the case of an accident is the reactor containment building. This four foot thick, reinforced concrete, steel lined building is shown in Figures 7 and 8. This barrier was partially breached for about the first four hours of the accident due to a failure of the building to isolate. Because of a design deficiency, the only isolation signal provided was a high pressure isolation at 4 psig which was not achieved until after substantial fuel cladding damage had occurred. However, due to the fact that most of the fission products which escaped the reactor building entered the auxiliary building, and since the exhaust ventilation system from this building passes through high efficiency particulate and iodine filters, the only fission products which escaped into the environment in substantial quantities were the noble gases.

In addition to these previously mentioned barriers, there are several safety related, high quality, redundant systems which are primarily designed to maintain the inventory in the reactor coolant system and keep the core cool in the event of any type of a loss of coolant accident.[11] Again looking at Figure 8, the most important of these systems are the high pressure injection/makeup system for small breaks where the pressure can be maintained, and the low pressure injection/decay heat system for larger breaks where the pressure rapidly drops. In addition to these active systems there are the core flood tanks which will passively inject water directly into the reactor vessel when the pressure goes below about 600 psig.

With this basic discussion of the design philosophy of a nuclear power plant as background and referring to Figure 8, the following is a very brief description of the major causes of the accident and its subsequent progression. (A detailed chronology of the first 16 hours of the accident before a stable condition was finally achieved is included as Appendix B).

At about 4:00 AM on Wednesday, March 28, 1979 the plant was operating normally at 97% power when both feedwater pumps tripped which in turn caused the turbine to trip. This trip is considered to be an anticipated transient which the plant was designed to handle with insignificant consequences. This sudden decrease in heat removal capability caused a very fast increase in pressure and temperature in the primary system. This in turn led to the opening of the power operated relief valve on the pressurizer followed very soon after by a reactor trip on high pressure. With the reactor trip the fission process in the core was stopped and the heat generation rate dropped to the decay heat rate, causing the pressure and temperature in the primary system to decrease. At this point the first unexpected problem occurred, the power operated relief valve failed to reclose. Unfortunately, the indication to the operator, which was only the electrical signal to the valve, indicated that it had reclosed. This mechanical failure in essence caused a small loss of coolant accident which was not recognized by the operator until much later into the sequence.

In addition to this mechanical failure, and as a result of an operational surveillance error, the emergency feedwater system, which started automatically upon the loss of normal feedwater, was blocked out by two valves which were closed in violation of the plant's technical specifications. This condition persisted for about 8 minutes until finally recognized by the operator after the steam generators had boiled dry. This temporary lack of feedwater to the steam generator by itself would not have led to the subsequent uncovering of the core. However, it did cause the transient to be much more severe, contributing to the misleading indications of pressurizer level. This level indication eventually led the operators to believe they had a full reactor coolant system and caused them to throttle back on the high pressure injection/ makeup pumps which had been injecting at full flow. Had these pumps been allowed by the operators to continue injecting full design flow, the decrease of inventory in the reactor coolant system would never have occurred. This operational error therefore was the primary cause of the eventual uncovering of the core.

Meanwhile, the water which was being relieved through the stuck open relief valve was filling the reactor coolant drain tank which eventually spilled its contents to the floor of the reactor building. Due primarily to the design deficiency of a lack of diverse signals for reactor building isolation, a significant amount of this water was automatically pumped over to tanks in the auxiliary building. This breach of containment, along with a suspected primary to secondary leak in one of the steam generators, was initially thought to be the primary release path of noble gases and possibly iodine from the plant. However, it was much later determined that the primary release path was normal and/or abnormal leakage through the letdown and makeup system and the gaseous radwaste system, the operation of which was required to maintain a stable cooling mode.

The loss of reactor coolant inventory, combined with insufficient makeup, continued for about the first 2 1/2 hours until finally an isolation valve upstream of the power operated relief was shut by the operator, terminating the loss of coolant accident. In the meantime, the operator had tripped all reactor coolant pumps due to excessive vibrations. This loss of forced reactor coolant flow, combined with the loss of coolant inventory, led to the uncovering and heatup of the core. The core was at least partially uncovered for about 1 1/2 hours until the power operated relief valve was isolated allowing the pressure in the system to increase above saturation. While the core was uncovered a zirconium metal-water reaction occurred which generated significant amounts of hydrogen and caused the release of significant amounts of fission products from the fuel rods. It is important to note that during this time, if the operators would have had sufficient indication to determine that the core was uncovered, they would have increased the high pressure injection/ makeup flow to full design flow. This would have quickly recovered the core preventing substantial fuel damage from occurring.

At about 6:40 AM several in-plant radiation monitors began to alarm, making it obvious that severe radiological problems were beginning to develop.

Based on this situation, Met Ed declared a site emergency and began to notify the appropriate offsite agencies according to their emergency plan. It was at this point that I first became involved in the accident. Being the Bureau's duty officer, at about 7:05 AM I was called by PEMA (Pennsylvania Emergency Management Agency) and informed that a site emergency had been declared and that I was to call the plant control room for technical details in accordance with our emergency plan. Upon calling the plant I was informed that they had suffered a small loss of coolant accident which had been terminated. They also told me that the plant conditions were now stable and no offsite releases were occurring. I then called the other key members of the Bureau and upon arriving in our office they established an open line with the control room at about 7:30 AM, again in accordance with our emergency plan. At about this time a general emergency was declared due to increasing radiation levels in the reactor building.

During the entire first few days we retained an open line with the plant control room. At all times we felt that Met Ed was being candid and giving us all the available information that they had on plant status and radiological monitoring. This information was being confirmed later that morning by NRC I & E personnel who arrived from the King of Prussia Office.

Also about this time we were informed by Met Ed that their initial dose assessment calculation indicated the possibility of a 10 rem/hr dose rate offsite near Goldsboro. This calculation was based on the radiation levels in the reactor building, and assumed a 50 psig pressure in the building (the actual pressure at this time was about 2-4 psig) and the release of a reference mix of radioisotopes. This immediately alerted us to the possibility of an evacuation and we called PEMA to alert York County. A few minutes later radiation surveys downwind of plant verified that no radiation levels above background were detectable. This, combined with the low pressure in the reactor building prompted us to call off this alert and the appropriate agencies were so notified.

By about 10:00 AM radiation levels in the range of 1-3 mrem/hr were first detected immediately offsite by the utility. This prompted us to send out a state monitoring team which verified the readings. For the remainder of Wednesday, surveys performed by teams from the state, utility, NRC and DOE confirmed that offsite levels of radiation were in the range of 1-10 mrem/hr (β-γ)[12] near the site. Occasionally higher levels were observed onsite, in the plume, and in relatively stagnant pockets.[13] This was primarily caused by the meterological conditions during the first few days of low wind speed and variable direction which resulted in very little dispersion.

Meanwhile at the plant, the operators were attempting various means of keeping the core cool and trying to establish a more stable cooling mode. There was sufficient evidence at this time to indicate that voids were present in the reactor coolant system and that significant fuel damage had occurred. The attempted methods varied from allowing the pressure to increase in order to collapse the voids and start a reactor coolant pump; to trying to depressurize in order to allow injection of the core flood tank in an attempt to assure the core was covered, and then trying to establish the normal cold shutdown cooling method using the decay heat removal system. Due mainly to the large amount of voiding in the reactor coolant system and the long period of time required to refill the system, these attempts were unsuccessful in establishing a stable cooling mode. However, they were successful in keeping the core covered and preventing further fuel damage.

Another event which occurred at about 2:00 PM, the possible significance of which went unnoticed or unrecognized by the operators, was a 28 psig pressure spike in the reactor building which is thought to be due to a localized hydrogen burn or explosion. The recognition of this event about a day and a half later led to an increased awareness on the part of the NRC in Washington to the possibility of further hydrogen or additional unknown problems.[14]

Finally, at about 8:00 PM Wednesday evening the operators were able to collapse the voids in the "A" loop and start a reactor coolant pump to establish forced circulation, thus finally establishing a stable cooling mode. It should be noted that although there was still a significant hydrogen/steam/noble gas void at the top of the reactor vessel, it was not interferring with the forced cooling and therefore this was a stable condition. In addition, because of the continuing operating of the letdown/makeup system, this gas void was slowly being reduced by dissolving in the reactor coolant system and being vented into the makeup tank.

After leaving the office a few hours earlier, I arrived back at about 8:00 AM on Thursday morning and immediately decided to go down to the site to get a clearer picture of how the situation was progressing. Upon arriving at the Observation Center, which is right across the river to the East of the plant, I interfaced directly with the Met Ed and NRC personnel who were there mainly coordinating the offsite monitoring effort. Throughout the day offsite radiation levels appeared to be trending downward with many stations approaching background levels. Average radiation levels downwind near the site were in the range of 1-3 mrem/hr with occasional higher levels onsite and directly in the plume.

While at the site on Thursday, I vividly remember seeing reports of radiation levels taken by helicopter above the plant vent as high as 3000 mrem/hr ($\beta - \gamma$). This is one of the major reasons why, on Friday morning when the 1200 mrem/hr ($\beta - \gamma$) reading above the vent was reported, we were not overly concerned about the eventually offsite doses or need for protective action.

Our major concern at this time was the need for locating the source of the releases and controlling them, which I expressed to Met Ed management and they concurred. I went home that evening feeling that the worst was over and all that remained was a very difficult clean-up operation. Little did I know that the next morning all hell was to break loose almost completely unnecessarily.

Shortly after arriving in the office at about 8:00 AM on Friday morning, we received information from the plant indicating that in the process of venting the makeup tank a release of noble gas had occurred. A helicopter which had been monitoring the release had detected a momentary level of 1200 mrem/hr ($\beta - \gamma$) about 150 feet directly above the plant vent. Utility and NRC monitoring teams downwind had detected maximum levels of about 20-25 mrem/hr ($\beta - \gamma$) immediately offsite near the Observation Center. These maximum levels were of very short duration and were decreasing rapidly to less than 1 mrem/hr. In addition, we had sent out a state monitoring team to perform surveys in the vicinity of the plant. They were also taking readings near the Observation Center and saw a maximum of about 17 mrem/hr (β - γ) for a short duration, essentially confirming the utility and NRC data. We were therefore confident that no protective action was required as a result of this release.[15]

About 9:00 AM, we received a notification from PEMA that they had received a telephone call from NRC headquarters in Washington recommending an immediate evacuation out to 10 miles, and were requesting our assessment of the situation. We told them that, based on the information that we had, there was no reason for any protective action, and that we would confirm our assessment and call them back. We immediately called NRC headquarters in Washington to find out the reason for their evacuation recommendation. I personally participated in the very frustrating conversation which followed. I informed them of our assessment of the situation, to which they did not seem to disagree or even take serious note. About all we could get out of them was that the recommendation was made by top management at NRC, the specific source of which they would not provide. After hanging up in frustration, we contacted our monitoring team and the plant to determine if the situation had changed significantly. After confirming the situation was stable and radiation levels were still decreasing, we attempted to call PEMA to confirm our initial assessment that no protective action was required. Unfortunately the local radio stations were already making announcements to prepare to evacuate. The excitement which was created by these announcements had completely overloaded the telephone system and we were not able to contact PEMA by phone. Therefore, it was decided that I should go to PEMA headquarters and Tom Gerusky, the director of our bureau, should go to the Governor's Office (both within reasonable walking distance) with the recommendation that no protective action be taken. In the meantime, Chairman Hendrie of the NRC from Washington had contacted Governor Thornburgh and had recommended a "take cover" within 10 miles of the plant, which was subsequently implemented.

Later that morning in another telephone conversation with the Governor, Chairman Hendrie, under the false assumption that substantial releases were occurring and were likely to continue in the future, stated almost matter of factly that if he had a pregnant wife and preschooler in the area, he would probably want them out. Thus came the recommendation for a precautionary advisory that pregnant women and children[16] leave the area within 5 miles of the plant. This advisory was later that morning given to the public by the Governor.

I was much later to learn firsthand from the people who were directly involved, the unfortunate series of misunderstandings that led to that Friday morning recommendation to evacuate. This event more than anything led to the escalation of a minor release into a full blown crisis, which continued for many days. A re-creation of those events are as follows:

Early Friday morning the plant operators, suspecting that leakage in the waste gas system was a major contributor to the release that were occurring, had been periodically shutting the vent on the makeup tank.[17] The pressure in the tank had slowly built up to the liquid relief setpoint and was relieving, thus threatening the normal recirculation mode of the makeup and reactor coolant pump seal water system.[18] The operators had decided to open the vent on the makeup tank to allow the continuation of this normal mode of operation. About an hour after the vent was opened, radiation levels of 1200 mrem/hr (β - γ) were measured from a helicopter about 150 feet above the plant vent. This was essentially the information that we received from the plant shortly after 8:00 AM.

Meanwhile, at the NRC Incident Response Center (IRC) in Bethesda, Md., an open line had been earlier established with the Unit 2 control room and they were being relayed information from an NRC I & E inspector. On Friday morning based on erroneous information, it was believed by the NRC in the IRC that the waste decay tanks were full. They therefore thought that the venting from the makeup tank was being compressed into the waste gas decay tanks, and these tanks were periodically relieving their contents at a discharge point downstream of the auxiliary building filters.[19]

Based on the above erroneous assumptions and using an assumed reactor coolant radioisotope concentration, NRC personnel in the IRC made a rough, conservative calculation which indicated that given these assumed circumstances the estimated offsite dose would be about 1200 mrem/hr. At about the same time this estimate was being given to the people in charge of the IRC,[20] the helicopter measurement of 1200 mrem/hr came in over the open line from the plant. Neglecting to verify the 1200 mrem/hr measurement and assuming it to be an offsite measurement, it was decided to recommend a downwind evacuation out to 10 miles. Unfortunately this recommendation was given directly to PEMA, completely bypassing our Bureau, which was supposed to have this responsibility. Fortunately it was never carried out.[21]

A short while later when the IRC finally realized that this 1200 mrem/hr level was directly above the plant vent, they performed another very conservative calculation which indicated that if this level persisted for a long period of time the offsite dose would be about 120 mrem/hr. This additional erroneous estimate, it is believed, then became the basis for Chairman Hendrie's recommendation to "take cover" 10 miles downwind.

The other major concern, which began on Friday and which probably caused even more unnecessary consternation than the misconceived evacuation, was the possibility of radiolysis[22] occurring in the reactor coolant system. It was first thought that the hydrogen and oxygen, which under certain conditions can be generated by radiolysis, was slowly increasing the size of the bubble in the reactor vessel thus eventually interferring with the forced cooling of the core and requiring the use of high pressure safety injection to keep the core covered. Later an even greater concern arose about the possibility of radiolysis. This had to do with the possible generation of oxygen having the potential to evenually cause an explosive gas mixture in the reactor vessel, which if detonated could have led to a core disruption accident.

As it turns out, all these concerns were completely groundless. It was not physically possible for any radiolysis to have occurred in the reactor coolant system due to the existence of a very large overpressure of hydrogen which totally inhibited this reaction.[23] In simple terms this means that the primary basis for all the speculation about possible core meltdown and precautionary evacuations which occurred over that first weekend did not even exist.

When these concerns about radiolysis first arose, most of the technical people involved, after careful consideration, did not believe that it presented a real problem. From my own experience with pressurized water reactors, I knew that a small excess concentration of hydrogen was maintained in the primary system to scavenge oxygen and prevent radiolysis. Most of the knowledgeable people that I discussed the problem with concurred that it was probably a very unrealistic assumption. However, there were a few NRC staff people who were perpetuating this concern. And unfortunately until the bubble was eventually dissipated by a deliberate venting of the reactor coolant system, this was considered to be the initiator of the worst case scenario for accident planning purposes.[24]

In my opinion, the reason for this error was that the people who were working on this problem in Washington were given the wrong assumptions concerning the conditions in the system. It would later be discovered that the radiolysis rate was calculated at atmospheric pressure,[25] while the real condition in the system was a pressure of about 1000 psig saturated with hydrogen.

It is unfortunate, but not surprising that the NRC would continue to use these most pessimistic and unrealistic assumptions in their discussions about possible scenarios. In my opinion, this was primarily due to the fact that the organization of the NRC was designed specifically to review and license nuclear power plants, in which they do a credible job. For this reason, they typically have groups of experts who review very specific areas. In this particular case, however, they were completely out of their element. These various groups of experts were typically predicting the worst in their particular area. Unfortunately, there were very few NRC personnel with a good overall working knowledge of the plant to sort out this sometimes conflicting and pessimistic information.

It is not surprising that these circumstances in turn led to obvious problems for the media in attempting to report the story. My first involvement with the media came early Wednesday morning while fielding questions at the first press conference. During this exchange I became painfully aware that much of the technical information the media was seeking was completely over their heads. This lack of technical knowledge which was evident throughout the entire episode, led to some misunderstandings and a tendency to get bogged down on minor details thus preventing the complete details from becoming known.

The other major factor which caused difficulty for the media was the many different sources of information during the first few days of the accident. These sources were typically giving similar information with varying degrees of pessimism. This situation understandably created a sense of confusion as to what was really happening.

Given all these shortcomings, the local media, especially the local radio stations, did an excellent job during the height of the crisis in sorting out the facts and getting accurate information to the public. Unfortunately, the national media generally tended to grossly sensationalize and distort what was actually happening and what the future might hold.[26] In the final analysis, the media must share some of the blame for creating the panic and crisis situation, a basis for which never existed to the degree that was reported.

It can be concluded that the TMI accident, although very serious, should not have caused by itself the crisis situation which existed for a very long period of time. The crisis was produced mainly by a combination of misinformation, poor communications and sensational media coverage.

Considering the number of successive operational, mechanical and design errors which caused the accident and the resulting fuel damage, the radiological consequences were relatively small. This can be considered fortunate because the lessons learned as a result of this accident have and will continue to improve the safety of nuclear power plants.

FOOTNOTES

1 A thermoluminescent dosimeter or TLD is a small beta-gamma dosimeter consisting of a semiconductor chip which records the cumulative amount of radiation received wherever the dosimeter has been placed. When a measurement is desired the dosimeter is placed in a reader which records the radiation damage to the semiconductor chip and then thermally anneals the chip to relieve the damage allowing reuse of the dosimeter.

2 As can be seen from these first two figures the Pennsylvania Bureau of Radiation Protection had a modest environmental monitoring program in effect prior to the accident, the primary purpose of which was to perform an independent check of Met Ed's more extensive monitoring program. The state monitoring program is currently in the process of being expanded around all nuclear power plants in Pennsylvania. This is the direct result of recently appropriated state funds which have been requested over the past several years for this purpose.

3 Noble gases as the name implies are chemically inert and therefore do not bioaccumulate in any organ. They are only a hazard mainly due to external gamma radiation as the cloud passes by. Radioiodine concentrates in the thyroid gland and also in cows milk and is therefore primarily an ingestion or inhalation problem.

4 Taking these considerations into account, a more likely maximum individual dose would be about 30 millirem due to noble gases.

5 The highest level of radioiodine found in milk was about 40 picocuries/liter for a short period of time. This is about a factor of 10 less than that found over a much wider area during the Chinese fallout episode of 1976.

6 It should be noted that this TLD data would have been the primary method of estimating population exposure. It is therefore unfortunate but not extremely important from the standpoint of determining population exposure that the plant vent monitors went off scale early into the accident.

7 Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station by Ad Hoc Population Assessment Group, May 10, 1979.

8 The person-rem concept is a means of measuring the collective dose received by a large population. It is simply determined by multiplying the dose to each segment of a population by the total number in that segment of the population. It is also a convenient method of determining risks to a population from exposure to radiation since most of the estimations are based on exposures to large population.

9 These risk factors are based on radiation exposures to entire average populations. They consequently take into consideration the risk to pregnant women and young children as well as others which are more susceptible to radiation exposure.

10 Fission products are a variety of radioactive elements which are created when the uranium atoms fission. In the process of decaying to a stable state they emit beta and/or gamma radiation. In this process they also generate heat, called decay heat, which must be removed even after the reactor has been shutdown to prevent the fuel from eventually melting. This decay heat level is about 6% of full thermal power immediately after reactor shutdown but decays very quickly following the exponential radioactive decay process of the fission products.

11 A loss of coolant accident is defined as any breach of the reactor coolant system, up to and including a double ended break of the largest pipe. This type of accident was considered to be the worst case design basis accident for a light water reactor. This philosophy will probably undergo substantial changes as a result of the lessons learned from the TMI accident.

12 These are open window measurements on portable survey meters which indicate the sum of the beta and gamma radiation. The much more penetrating gamma radiation was also routinely measured by closing the windows and was typically about 1/3 to 1/5 of this total beta/gamma measurement.

13 The maximum recorded reading offsite was 70 mrem/hr (β - γ) near the North gate for a short period of time.

14 This undoubtedly led to an increased anxiety on the part of the NRC that the accident was much more severe than originally thought, and probably set the stage for the misconceived evacuation recommendation on Friday morning.

15 Later data was to indicate that this release on Friday morning, which caused the ensuing anxiety and precipitous actions actually delivered only a few percent of the total dose received by any member of the public during the entire duration of the accident. Based on the monitoring information we had received throughout the course of the accident, we felt confident that the maximum cumulative offsite dose to any individual was less than 100 millirem. This was a factor of ten less than the EPA protective action guidelines upon which our plan was based and was consequently where we would have been prepared to recommend protective action to limit further public exposure.

16 Tom Gerusky, who was in the Governor's Office at this point, did not recommend against this advisory primarily because it was precautionary. It was thought that NRC should have been more knowledgeable about the real situation, and if our information was in error, it would have been very difficult to justify not taking this conservative course of action.

17 The makeup tank is essentially the surge tank for the reactor coolant letdown and makeup system, which was required at this time for the continued operation of the reactor coolant pump without unnecessarily drawing down the emergency supply of borated water. This tank is normally vented to the waste gas header which was suspected of having a leak, and which was causing a periodic release to the environment through the filtered auxiliary building ventilation system.

18 The source of makeup water could have been switched to the emergency borated water storage tank. However, this would have led to the eventual depletion of this tank and the consequent need to recirculate the reactor building sump water. Using this relatively unpurified source could have eventually led to more severe operational problems, and therefore the normal letdown/makeup system lineup was the preferred mode.

19 Actually, the waste decay tanks at this time were only at about 2/3 of their design pressure, but this had been a concern of the utility and they were in the process of rigging up a temporary line to vent these tanks into the reactor building.

20 Harold Denton was the senior NRC type in the IRC that morning and it was primarily his decision to recommend an evacuation.

21 Harold Denton was later to say that his concerns about evacuation went down by orders of magnitudes once he arrived on site later that afternoon and became better appraised of the situation.

22 Radiolysis in the decomposition of water into hydrogen and oxygen due to interaction of intense neutron and gamma irradiation.

23 In borated water solutions the rate of radiolytic decomposition is directly proportional to the energy absorption from neutron scattering and capture minus the gamma energy absorption. (Ref: Etherington, Nuclear Engineering Handbook, 1st Edition, 1958, p. 10-132). In addition, in gamma and neutron fields typical of power reactors, a hydrogen concentration of only 17 cc/kg is needed to suppress radiolysis in the primary coolant. (Ref: US Patent 2937981, 5/24/60). Noting that after the control rods were inserted the neutron flux was reduced by many orders of magnitude and that the actual hydrogen concentration in the reactor coolant on Friday was about 1670 cc/kg, it is obvious that radiolysis in the reactor coolant system was not physically possible.

24 The worst case scenario that was speculated was a core meltdown. According to the results of WASH-1400, the most exhaustive and authoritive study on the subject, the following would be the consequences of a reactor core meltdown. (No fault was found with this consequence model in the recent highly publicized independent review of this report.) The most likely core melt sequence (about 90% of all the possible scenarios leading to core melt) would be a core melt through, with the molten core eventually penetrating the base of the containment building and solidifying a few tens of feet beneath. The most likely conseqeunces of this sequence would be very small; less than one early fatality, less than one additional latent cancer fatality per year and less than one additional genetic effect per year. (In the case of TMI, there would have been substantial groundwater and possible river water contamination that would have been difficult to clean up.)
Prior to melt through there is the additional risk, based partly on the availability of some additional safeguard equipment, that the containment vessel could be breached. Assuming the worst possible atmospheric breach of containment combined with the worst case meterology, population distribution and evacuation scenario, the maximum possible consequences would be much more serious. This could include about 3000 early fatalities, a 9% increase in fatal cancers, and a 2% increase in genetic effects to the assumed population.

In addition, there could be the requirement for temporary relocation from an area of about 300 square miles (much of which could be reclaimed in a short period of time with minimal decontamination) and crop and milk restrictions within an area of about 3000 square miles.

25 This was the condition in the reactor building outside of the reactor coolant system. Radiolysis was probably occurring in the water that was spilled on the floor of this building. This was one of the reasons for wanting to get a hydrogen recombiner in operation as soon as possible. The maximum hydrogen concentration in this building was measured at about 2.2%, well below the 4% necessary for burning or the 8% necessary for explosion.

26 It seemed the further away one went from TMI the worse the situation was reported as being. In fact, some foreign media reported that thousands had died as a result of the accident.

TABLE 1

Comparison of TMI Accident Dose with Natural Background Radiation

Estimates of natural background radiation levels at various locations in the U.S.

	Annual Dose Rate mrem/yr			
Location	Cosmic Radiation	Terrestrial Radiation	Internal Radiation	Total
Atlanta, Georgia	44.7	57.2	28	130
Denver, Colorado	74.9	89.7	28	193
Las Vegas, Nevada	49.6	19.9	28	98
Harrisburg, PA	42.0	45.6	28	116

Living in Denver, Colorado compared to Harrisburg, PA - + 80 mrem/yr

Living in a brick house instead of a wood frame house - + 14 mrem/yr

Variation in natural background radiation within 50 miles of TMI - 100 to 130 mrem/yr

Radiation dose delivered as a result of TMI accident

Individuals remaining out of doors at location of highest estimated offsite dose - less than 100 mrem

Average dose to a typical individual within 50 miles of the site - 1.5 mrem

10 miles of the site - 8 mrem

Population dose within 50 miles of site - 3300 person-rem

Natural background radiation dose during the same 11 day period above

Average background dose to typical individual - 3.5 mrem

Population background dose within 50 miles - 7500 person-rem

TABLE 2

Risk Factors for Low-Level Radiation Exposure

RADIATION-INDUCED CANCER MORTALITY ESTIMATED IN THE 1972 BEIR REPORT (3)

	1972 BEIR Report Estimates: Annual number of deaths resulting from exposure of the U.S. population to a radiation dose rate of 0.1 rem [100 millirem] per year[(a)]		Derived Risk: Number of Cancer Deaths per 10^6 person-rem[(b)]	
	Absolute Risk Model	Relative Risk Model	Absolute Risk Model	Relative Risk Model
Leukemia	516	738	26	37
Other Fatal Cancers				
Assumption A:[(c)]	1210	2436	61	123
Assumption B:[(d)]	1485	8340	75	421
Total (Range)[(e)]	1726-2001	3174-9078	87-101	160-458
Nominal Range[(f)]	1700-2000	3200-9100	90-100	160-460
		Geometric mean $(95 \times 310)^{1/2}$	=	200 (172)

(a) 1967 U.S. population = 197,863,000. Collective Dose Rate = $(198 \times 10^6$ people) x (0.1 rem/yr) = 19.8×10^6 person-rem/year. From Table 3-3 (Relative Risk and Table 3-4 (Absolute Risk) of the 1972 BEIR Report (3) pp. 172-173.
(b) 1972 BEIR Values (Cancer deaths/year) divided by the collective dose rate of 19.8 [10^6 person-rem]/year.
(c) Assumption A: 30-year period of elevated risk following irradiation.
(d) Assumption B: Lifetime period of elevated risk following irradiation.
(e) Low estimate = Leukemia Risk + Assumption A for other fatal cancers.
High estimate = Leukemia Risk + Assumption B for other fatal cancers.
(f) Preceeding values rounded to two significant figures.

ESTIMATES OF GENETIC EFFECTS OF LOW-LEVEL IONIZING RADIATION

Disease Classification	Natural Incidence (per 10^6 live births)	Effects per 10^6 live births[(a)] of 5 rem per generation[(b)]		Estimated Risk per 10^6 person-rem[(c)]	
		First Generation	Equilibrium	First Generation	Equilibrium
Dominant diseases	10,000	50 to 500	250 to 2500	6 to 60	30 to 300
Chromosomal and recessive diseases	10,000	relatively slight	very slow increase	relatively slight	very slow increase
Congenital anomalies	15,000				
Anomalies expressed later	10,000	5 to 500	50 to 5,000	0.6 to 60	6 to 600
Constitutional and degenerative diseases	15,000				
TOTAL	60,000	60 to 1000	300 to 7500	7 to 120	36 to 900
Risk per 10^6 people	1,200[(d)]/year				
			Geometric Mean	$(36 \times 900)^{1/2}$	= 200 (180)

(a) From the 1972 BEIR Report (3), Table 4 p. 57 which is believed to be erroneously titled. This table, like the preceding tables 2-3 pp. 54-55, is believed to be for a population of one million "live births" not for a population of one million. The range of values corresponds to assumed coubling doses between 20 rem (high values) and 200 rem (lower values).

(b) A generation is assumed to be 30 years.

(c) Risk per 10^6 person-rem = (cases/10^6 live births) x (30 years/5 rem) x (4×10^6 live births/year per 2×10^8 people) = 0.12 x cases/10^6 live births.

(d) Cases/10^6 live births x (4×10^6 live births per year/ 2×10^8 people).

TABLE 3

PROJECTED POTENTIAL HEALTH IMPACT OF THE THREE MILE ISLAND ACCIDENT TO THE OFFSITE POPULATION WITHIN 50 MILES

Effect	Estimated Number who would normally develop effect	Potential Impact of Natural Background Radiation	Potential Lifetime Impact of Population Dose from the TMI Accident from March 28, 1979 through April 7, 1979	
			Range[a]	Central Estimate[b]
Fatal Cancers	325,000[c]	1,700 - 9,000[d]	0.15 - 2.4[e]	0.7
Non-Fatal Cancers	216,000[f]	1,700 - 9,000[d,g]	0.15 - 2.4[c,f]	0.7
Genetic Effects first generation	78,000[h]	60 - 970[i]	(0.01 - 0.64)[j]	-
all future generations	-		0.05 - 4.8[k]	0.7
	All Health Effects		0.4 - 10[l]	2.0[l]

Footnotes

(a) This represents the extreme range of health effects estimates considering both the range of the collective dose estimates and the range of the estimates of the risks of low-level ionizing radiation as estimated in the 1972 BEIR Report (3).

(b) The central estimate is based upon taking the geometric mean (square root of the product) of the upper and lower bounds of the dose-to-health-risk conversion factors from Table 4-1 and multiplying this by the mean estimate of the population dose (3,300).

(c) Based upon the American Cancer Society projection that the risk of cancer death is 0.15 (0.15 x 2,164,000 = 324,600).

(d) Based upon multiplying the annual rates in Table 4-7 by 70 years, the mean life span.

(e) Based upon multiplying the lower range estimate of the population dose (1,600 person-rem) by the lower range of the absolute radiation-induced cancer risk (90×10^{-6}) and the upper range estimate of the population dose (5,300) by upper range of the relative radiation-induced cancer risk (460×10^{-6}).

(f) Based upon the difference between the American Cancer Society projection of the risk of getting cancer (0.25) and the risk of dying of cancer (0.15). The value given is the product of this difference (0.25 - 0.15 = 0.10) and the size of the population (2,164,000).

(g) Based upon the assumption that there are twice as many cancers as there are cancer fatalities.

(h) Based upon the natural annual incidence of genetic effects (1,200 per year per 10^6 population) from table 4-2 times an assumed reproductive period of 30 years.

(i) Based upon multiplying the risk to the first generation from table 4-2 by an assumed reproductive period of 30 years and by the natural background dose rate of 270,500 person-rem per year.

(j) Based upon multiplying the lower bound of first generation risk (7×10^{-6}) from Table 4-2 by the lower bound of the collective dose estimate (1,600 person-rem) and multiplying the upper bound of the first generation risk (120×10^{-6}) from Table 4-2 by the upper bound of the collective dose estimate (5,300 person-rem). The first generation risk is included in the risk to all generations and therefore, should not be separately added into the total.

(k) Based upon the procedure described in (j) but using the equilibrium risk bounds rather than the first generation risk.

(l) This is done for the convenience of providing an estimate of the total potential health impact. Technically, the effects are not equivalent and cannot be added.

Met-Ed Milk Sample
Met-Ed + PA Milk Sample
Met-Ed Air Sampler
PA Air Sampler

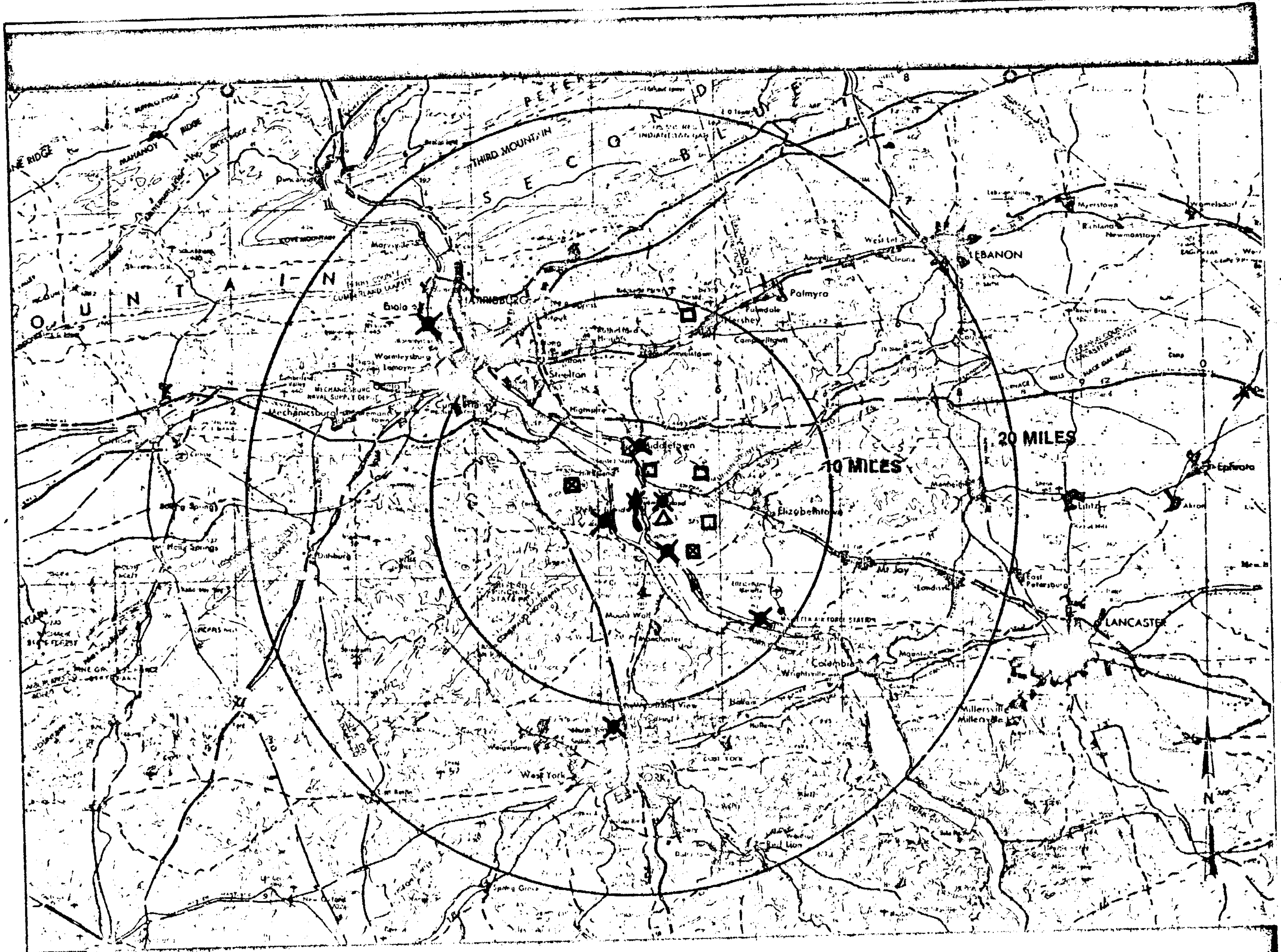


FIGURE 1: Air and Milk Sample Locations Prior to Accident

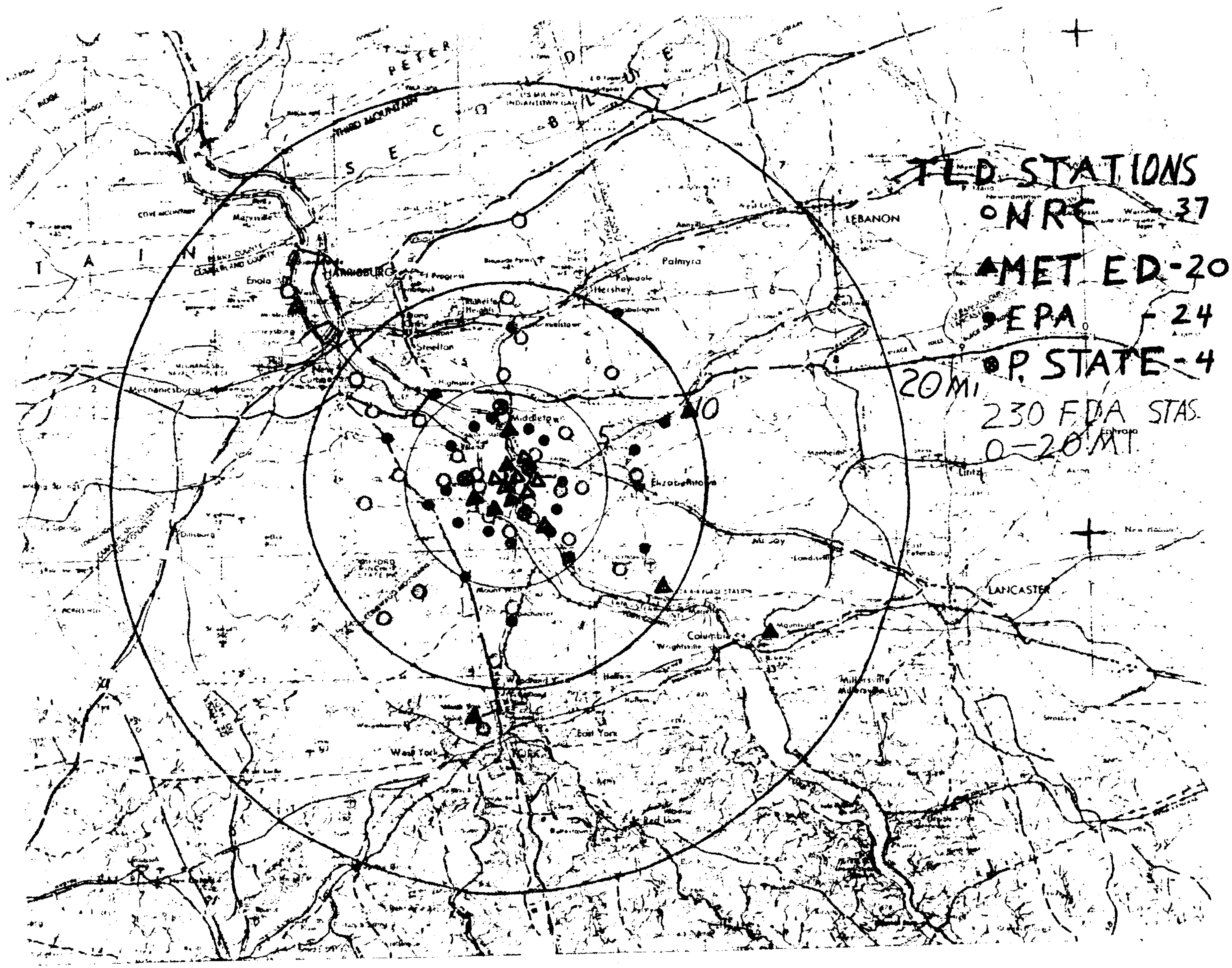


FIGURE 2: TLD Stations

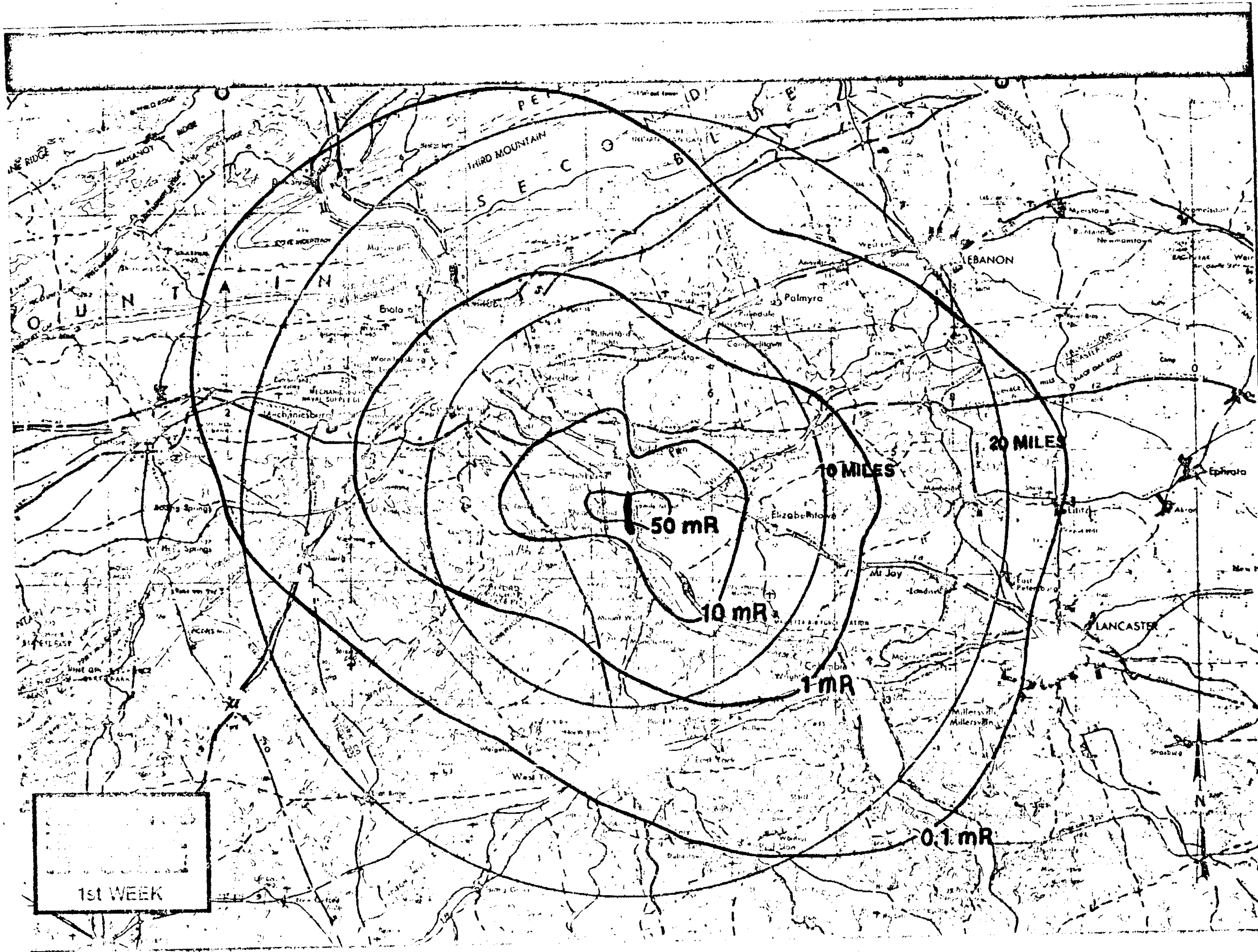
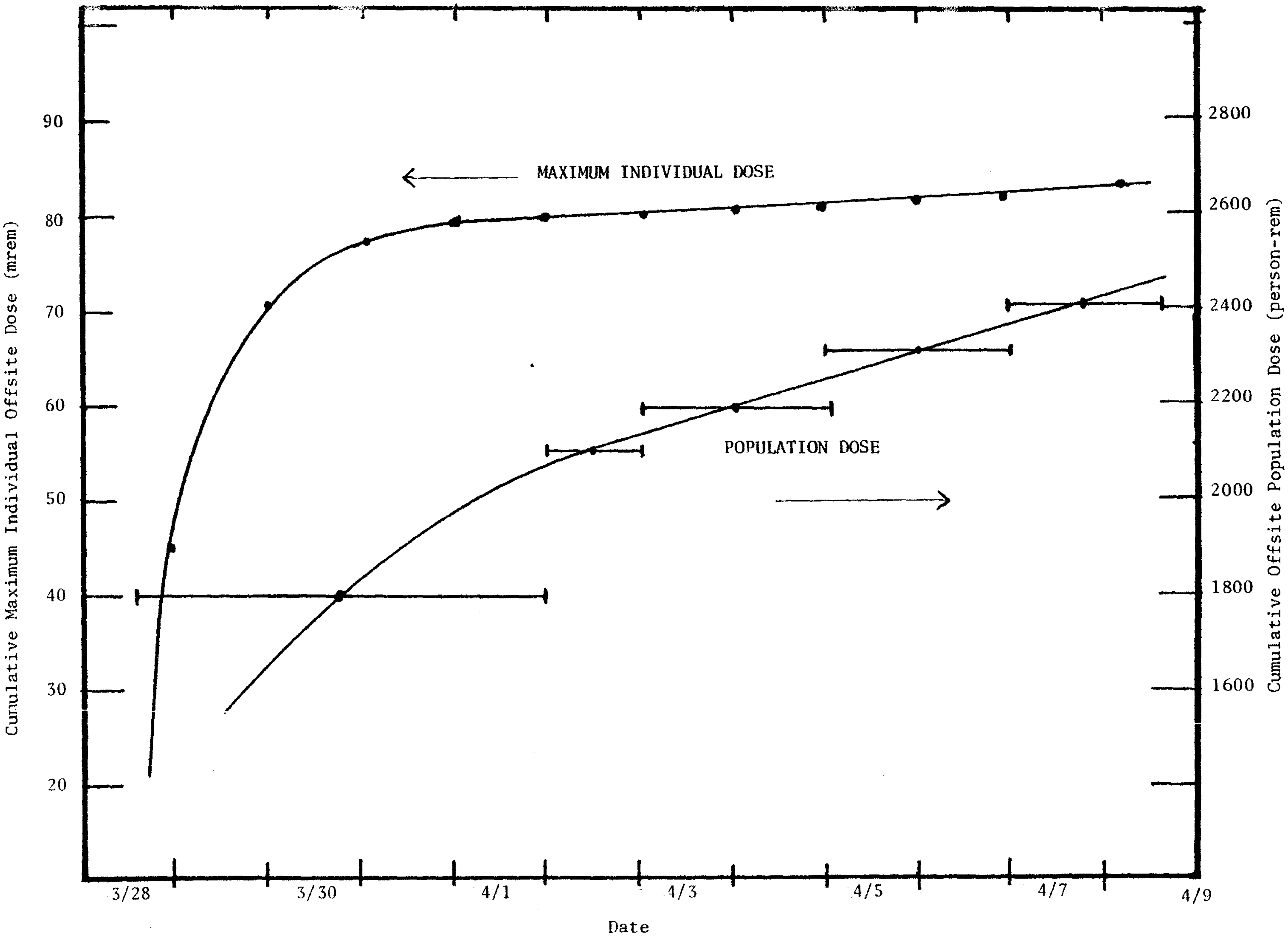
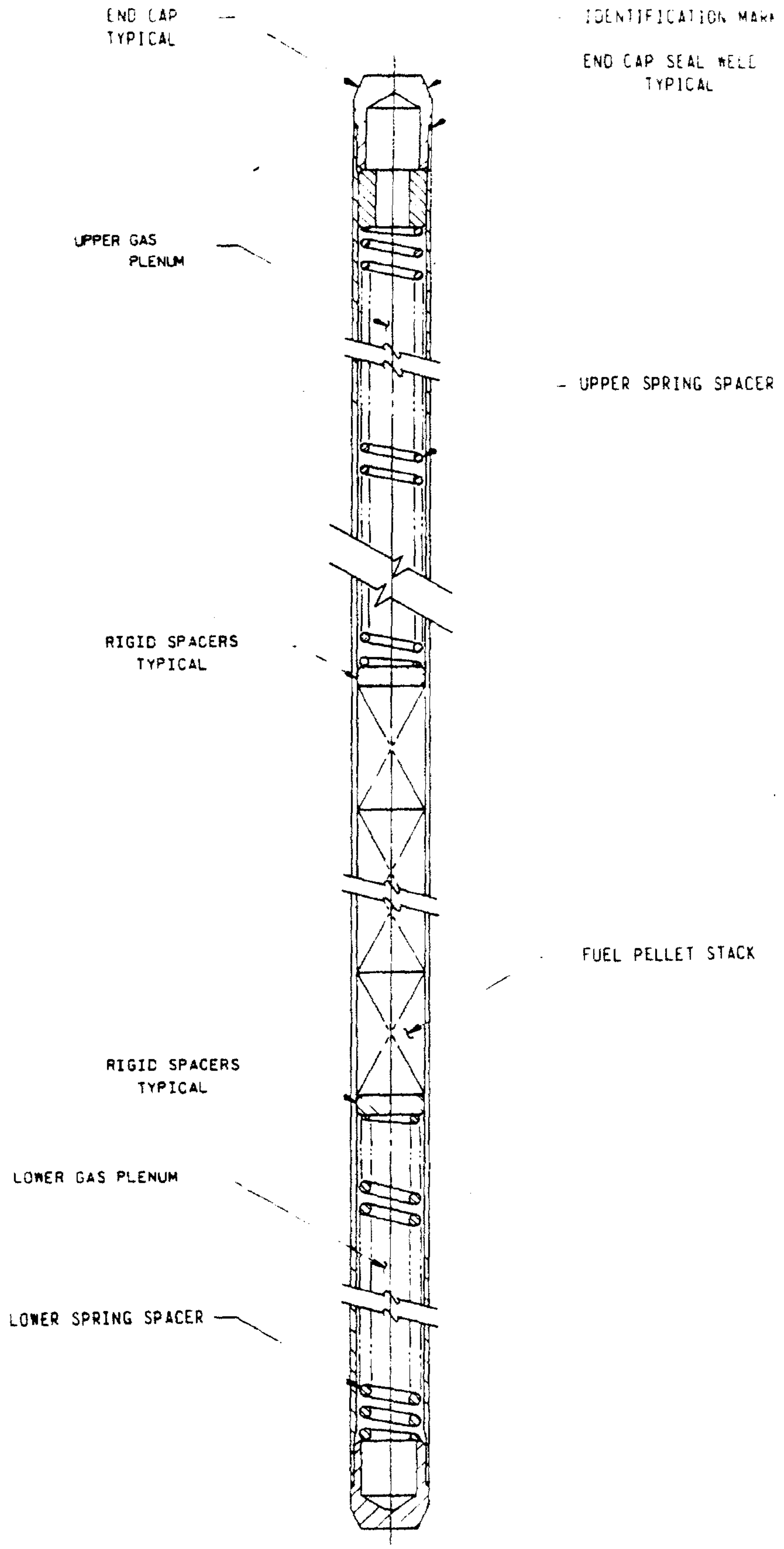


FIGURE 3: Estimated External Exposure for the First Week Following the Accident from TLD Data

FIGURE 4. Offsite Maximum Individual and Population Dose from Noble Gas
(NRC Model)





PREPRESSURIZED FUEL ROD

THREE MILE ISLAND NUCLEAR STATION UNI

FIGURE 5

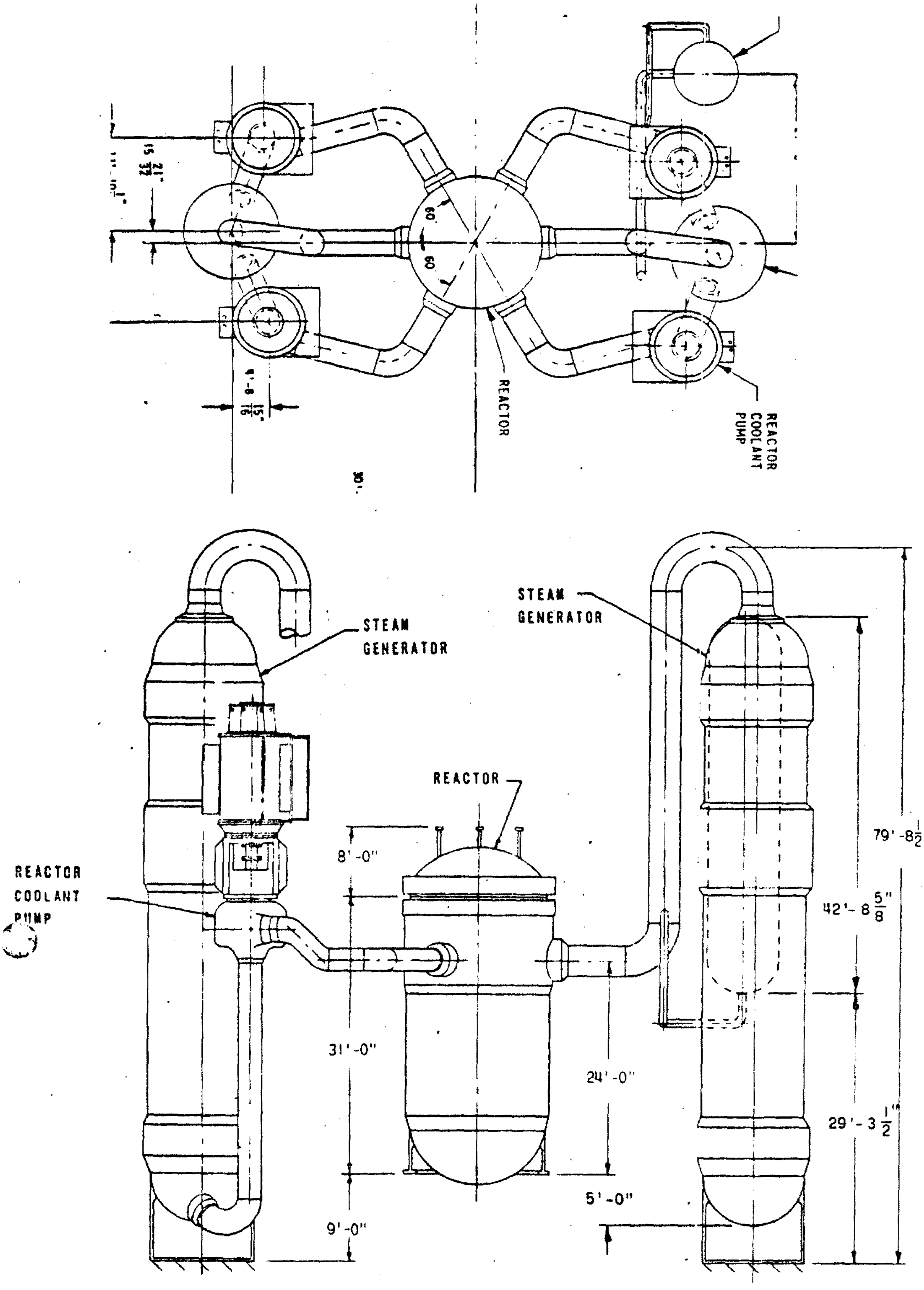


FIGURE 6: Elevation and Plan View of the TMI-2 Reactor Coolant System

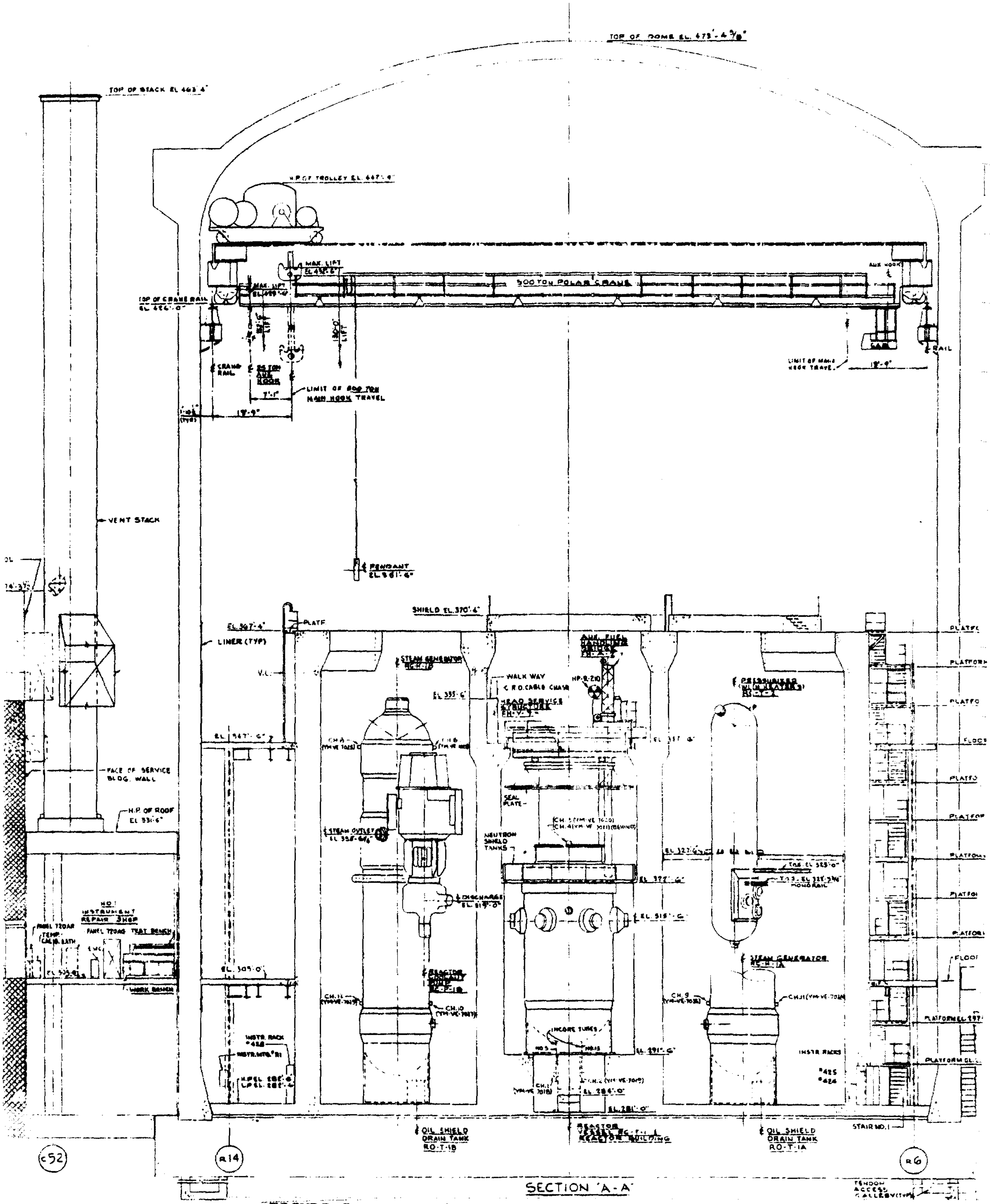


FIGURE 7: Elevation view of the layout of the TMI-2 Reactor Building

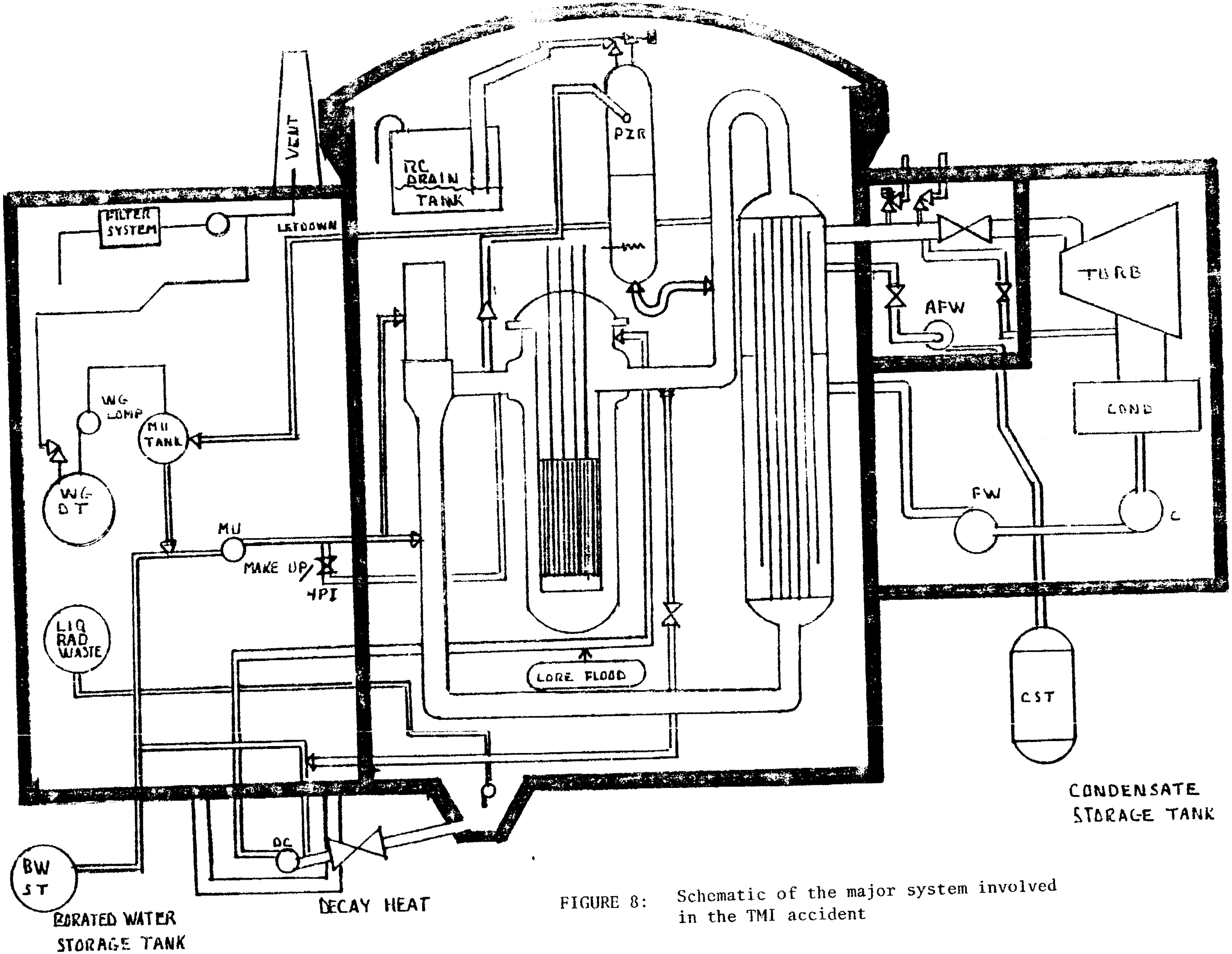


FIGURE 8: Schematic of the major system involved in the TMI accident

SUMMARY and DISCUSSION of FINDINGS from:

POPULATION DOSE and HEALTH IMPACT of the ACCIDENT at the THREE MILE ISLAND NUCLEAR STATION

(A preliminary assessment for the period March 28 through April 7, 1979)

Ad Hoc Population Dose Assessment Group

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May 10, 1979

This document contains only the "Preface" and "Summary and Discussion of Findings" sections of the full report. If the complete report is required, it may be obtained by calling (301) 443-3434, or writing to:

HFX-25, Bureau of Radiological Health
5600 Fishers Lane
Rockville, MD 20857

PREFACE

This report was prepared by technical staff members of the Nuclear Regulatory Commission (NRC), the Department of Health, Education and Welfare (HEW), and the Environmental Protection Agency (EPA), who constitute an Ad Hoc Population Dose Assessment Group. It is an assessment of the health impact on the approximately 2 million offsite residents within 50 miles of the Three Mile Island Nuclear Station from the dose received by the entire population (collective dose). The Ad Hoc Group has examined in detail the available data for the period up to and including April 7, 1979. Based on a preliminary review of data from periods beyond April 7, it appears that the collective dose will not be significantly increased by extending the period past April 7.

The dose and health effects estimates are based primarily on thermoluminescent dosimeters placed at specific onsite and offsite locations. The dosimeters measure the cumulative radiation exposure that occurred at these locations. They permit the most direct evaluation of dose to the offsite population from radionuclides (radioactive materials) released to the environment.

The report also addresses several areas of concern about the types of radionuclides released, about the contribution to population exposure due to beta radiation (which does not penetrate the clothing and skin) emitted from the released radionuclides, about the degree of coverage afforded by available radiation measurements, and about the range of health effects that may result from the estimated collective dose.

Based on the current assessment, the Ad Hoc Group concludes that the offsite collective dose associated with radioactive material released during the period of March 28 to April 7, 1979 represents minimal risks (that is, a very small number) of additional health effects to the offsite population. The numerical statement of this conclusion is developed in the report. The Ad Hoc Group is not aware of any radiation measurements made during this period that would alter this basic conclusion, although refinement of the numerical estimates can be expected as the data are updated and verified. The members of the Ad Hoc Group concur that the manner in which the collective dose estimates were computed was conservative (overestimated the actual dose). The uncertainties in the collective cose estimates and health effects are not large enough to alter the Group's basic conclusion, that is, the risk is minimal.

POPULATION DOSE AND HEALTH IMPACT OF THE ACCIDENT AT THE THREE MILE ISLAND NUCLEAR STATION

(A preliminary assessment for the period March 28 through April 7, 1979)

Summary and Discussion of Findings

An interagency team from the Nuclear Regulatory Commission (NRC), the Department of Health, Education and Welfare (HEW) and the Environmental Protection Agency (EPA) has estimated the collective radiation dose received by the approximately 2 million people residing within 50 miles of the Three Mile Island Nuclear Station resulting from the accident of March 28, 1979. The estimates are for the period from March 28 through April 7, 1979, during which releases occurred that resulted in exposure to the offsite population. The principal dose estimate is based upon ground-level radiation measurements from thermoluminescent dosimeters located within 15 miles of the site. These estimates assume that the accumulated exposure recorded by the dosimeters was from gamma radiation (that is, penetrating radiation that contributes dose to the internal body organs). The data were obtained from dosimeters placed by Metropolitan Edison Company before the accident (as part of their normal environmental surveillance program), from dosimeters placed by Metropolitan Edison after the accident and covering the period to April 6, and from dosimeters placed by NRC from noon of March 31 through the afternoon of April 7, 1979. These measurement programs are continuing. The results for the period beyond April 7, 1979 have not been fully examined. An additional dose estimate developed by the Department of Energy using aerial monitoring that commenced about 4 p.m. on March 28, 1979 is also included. A variety of other data helpful in assessing relatively minor components of collective dose was also reviewed.

The collective dose to the total population within a 50-mile radius of the plant has been estimated to be 3300 person-rem. This is an average of four separate estimates that are 1600, 2800, 3300, and 5300 person-rem. The range of the collective dose values is due to different methods of extrapolating from the limited number of dosimeter measurements. An estimate provided by the Department of Energy (2000 person-rem) also falls within this range. The average dose to an individual in this population is 1.5 mrem (using the 3300 person-rem average value).

The projected number of excess fatal cancers due to the accident that could occur over the remaining lifetime of the population within 50 miles is approximately one. Had the accident not occurred, the number of fatal cancers that would be normally expected in a population of this size over its remaining lifetime is estimated to be 325,000. The projected total number of excess health effects, including all cases of cancer (fatal and non-fatal) and genetic ill health to all future generations, is approximately two.

These health-effects estimates were derived from central risk estimates within the ranges presented in the 1972 report of the Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR) of the National Academy of Sciences. Preliminary information on the recently updated version of this report indicates that these estimates will not be significantly changed.

It should be noted that there exist a few members of the scientific community who believe the risk factors may be as much as two to ten times greater than the estimates of the 1972 BEIR report. There also is a minority of the scientific community who believe that the estimates in the 1972 BEIR report are two to ten times larger than they should be for low doses of gamma and beta radiation.

The maximum dose that an individual located offsite in a populated area might receive is less than 100 mrem. This estimate is based on the cumulative dose (83 mrem) recorded by an offsite dosimeter at 0.5 mile east-northeast of the site and assumes that the individual remained outdoors at that location for the entire period from March 28 through April 7. The estimated dose applies only to individuals in the immediate vicinity of the dosimeter site. The potential risk of fatal cancer to an individual receiving a dose of 100 mrem is about 1 in 50,000. This should be compared to the normal risk to that individual of fatal cancer from all causes of about 1 in 7.

An individual was identified who had been on an island (Hill Island) 1.1 miles north-northwest of the site during a part of the period of higher exposure. The best estimate of the dose to this individual for the 10-hour period he was on Hill Island (March 28 and March 29) is 37 mrem.

A number of questions concerning this analysis are posed and briefly answered below. More detailed discussions are included in the body of the report.

<u>What radionuclides were in the environment?</u>

The principal radionuclides released to the environment were the radioactive xenons and some iodine-131. Measurements made by the Department of Energy in the environment, measurement of the contents of the waste gas tanks, of the gases in the containment building and the actual gas released to the environment confirmed that the principal radionuclide released was xenon-133. Xenon-133 is a noble gas (which is chemically non-reactive) and does not persist in the environment after it disperses in the air. It has a short half-life of 5.3 days and produces both gamma and beta radiation. The risk to people from xenon-133 is primarily from external exposure to the gamma radiation, which penetrates the body and exposes the internal organs.

<u>What were the highest radiation exposures measured outside the plant buildings?</u>

Some of the Metropolitan Edison dosimeters located on or near the Three Mile Island Nuclear Station site during the first day of the accident recorded net cumulative doses as high as 1020 mrem. These recorded exposure readings do not apply directly to individuals located offsite. However, the onsite dosimeter readings were included in the procedure for projecting doses to the offsite population. This procedure is described in the report.

<u>What is meant by collective dose (person-rem)?</u>

The collective dose is a measure of the total radiation dose which was received by the entire population within a 50-mile radius of the Three Mile Island site. It is obtained by multiplying the number of people in a given area by the dose estimated for that area and adding all these contributions.

Were the radiation measurements adequate to determine population health effects?

The extensive environmental monitoring and food sampling were adequate to characterize the nature of the radionuclides released and the concentrations of radionuclides in those media. The measurements performed by Department of Energy (aerial survey) and Metropolitan Edison and Nuclear Regulatory Commission (ground level dosimeters) are sufficient to characterize the magnitude of the collective dose and therefore the long-term health effects. However, a single precise value for the collective dose cannot be assigned because of the limited number of fixed ground level dosimeters deployed during the accident.

How conservative were the collective dose estimates?

In projecting the collective dose from the thermoluminescent dosimeter exposures, several simplifying assumptions were made that ignored factors that are known to reduce exposure. In each case, these assumptions introduced significant overestimates of actual doses to the population. This was done to ensure that the estimates erred on the high side. The three main factors that fall into this category are:

(1) No reduction was made to account for shielding by buildings when people remained indoors.

(2) No reduction was made to account for the population known to have relocated from areas close to the nuclear power plant site as recommended by the Governor of Pennsylvania, or who otherwise left the area.

(3) No reduction was made to account for the fact that the actual dose absorbed by the internal body organs is less than the dose assumed using the net dosimeter exposure.

What is the contribution of beta radiation to the total dose?

Beta radiation contributes to radiation dose by inhalation and skin absorption. The total beta plus gamma radiation dose to the skin from xenon-133 is estimated to be about 4 times the dose to the internal body organs from gamma radiation. This additional skin dose could result in a small increase in the total potential health effects (about 0.2 health effect) due to skin cancer. The increase in total fatal cancers over that estimated for external exposure from gamma radiation alone would be about 0.01 fatal skin cancer. This contribution would be considerably decreased by clothing. The dose to the lungs from inhalation of xenon-133 for both beta and gamma radiation increases the dose to the lungs by 6 percent over that received by external exposure.

What radionuclides were found in milk and food and what are their significance?

Iodine-131 was detected in milk samples during the period March 31 through April 4. The maximum concentration measured in milk (41 pCi/liter in goat's milk, 36 pCi/liter in cow's milk) was 300 times lower than the level at which the Food and Drug Administration (FDA) would recommend that cows be removed from contaminated pasture. Cesium-137 was also detected in milk, but at concentrations expected from residual fallout from previous atmospheric weapons testing. No reactor-produced radioactivity has been found in any of the 377 food samples collected between March 29 and April 30 by the FDA.

Why have the estimates of radiation dose changed?

The original Ad Hoc Group estimate of collective dose (1800 person-rem presented on April 4 at the hearings before the Senate Subcommittee on Health and Scientific Research covered the period from March 28 through April 2. The data used for this estimate were obtained from preliminary results for Metropolitan Edison offsite dosimeters for the period March 28 through March 31 and preliminary results for NRC dosimeters for April 1 and 2. On April 10, the estimate of 2500 person-rem presented to the Senate Subcommittee on Nuclear Regulation by NRC Chairman Hendrie included the time period from March 28 through April 7. The data base for this estimate included additional NRC dosimetry results for April 3 through 7. The Ad Hoc Group's preliminary report of April 15 stated a value of 3500 person-rem for the time period from March 28 through April 7. This value resulted from better information on the dosimeter measurements and an improved procedure for analyzing the measurements.

The current report states an average value of 3300 person-rem (with a range of 1600 to 5300 person-rem) for the time period from March 28 through April 7. Additional dosimeter data were available and better methods were used to determine the collective dose. Also, the onsite dosimeter measurements are all included in the analysis.

The original estimate of maximum dose (80 mrem) to an individual presented on April 4 increased to 85 mrem in the April 15 preliminary report as a consequence of adding the contribution from April 2 to April 7. This estimate has now been revised slightly to 83 mrem, which is presented as less than 100 mrem so as not to imply more precision than this estimate warrants. New information on dosimeter readings on or very near the site was received after the initial analysis. It was also learned that an individual was present on one of the nearby islands (Hill Island) for a total of 10 hours during the period March 28 to March 29. The best estimate of the dose which may have been received by the individual is 37 mrem. The text includes a range of dose estimates for that individual.

Will these estimates of dose change again?

The dose and health effects estimates contained in this report are based on the dosimeter results for the period March 28 to April 7, 1979. There still remain some questions concerning interpretation of the dosimeter results. For example, the best values for subtracting background from the Nuclear Regulatory Commission dosimeters have not been determined. Recently available data from additional dosimeters exposed during the March 28 to April 7 period have been reviewed briefly, but could not be included in the calculations in time for this report. The actual contribution to collective dose from the period after April 7, if any, has not been fully assessed. Therefore, the numerical dose values may be subject to some modification.

The Ad Hoc Group feels that these factors represent only minor corrections to the present estimates. In any case, none of the above refinements should cause an increase in any of the current estimates that would alter the basic conclusion regarding the health impact due to the Three Mile Island accident.

Chronology of TMI-2 Accident 3/28/79

	Events
t =-1 sec. (0400:36)	Plant operating normally (2155 psig) at 97%. Cond. polisher valve closed due to malfunction in air system. Booster pumps (2 of 3 operating) may have been first to trip. One condensate pump tripped (2 of 3 operating). Loss of both feedwater pumps on low suction pressure. Turbine trip.
t = 0 +	All three emergency feedwater pumps started (operating pressure at t = 14 sec.)
t = 3 sec.	E-M relief valve open at 2255
t = 8 sec.	Reactor trip on high pressure at 2345
t = 13 sec.	Operator isolated letdown, started another MU pump and opened HP injection isolation valve in anticipation of expected pressurizer level decrease.
t = 13 sec.	E-M relief valve solenoid de-energized giving closed position indication at 2205 psi (Valve did not reseat)
t ≈ 10 sec.	RCS temp. peaks at 611° F, 2345 psi pressurizer level peaks at 255 inches.
t = 38 sec.	Emergency feedwater valves open on S/G low level. Block valves closed so no feedwater admitted. S/G boil dry at t = 1:45. Pressure indication and valve position is only indication operator had of system status.
t = 1-4 min.	Pressurizer level started increasing. Based on rate of increase being greater than rate that can be accounted for, it is suspected that one or more steam voids formed in RCS at this time. This was the first indication, along with the still increasing pressure in the RC drain tank, which the operator had that would indicate a departure from what would normally be expected. Normally level and pressure would trend together following a loss of feedwater transient. Departure from normal was due to EM relief valve being open causing a reduction in pressure, while the loss of heat sink (S/G's boiling dry) was causing an expansion of the RCS. It is suspected that level instruments were not greatly in error based on an evaluation of all conceivable types of malfunctions.

t = 2:04 min.	ECCS (HPI) initiation at 1600 psi.
t = 2:12 min.	RC drain tank relief valve lifted. RC drain tank high temp. alarm at t = 3:26 min. Further indication of open E-M relief valve.
t = 3:14 min.	Operator bypassed HPI portion of ECCS and throttled one of two injection isolation valves on "A" MU pumps in attempt to control pressurizing level. This reduced MU flow rate to about 3/4 of full flow at this operating point.
t = 4:38 min.	Operator tripped MU pump "C" in further attempt to control pressurizing level. This reduced MU flow rate to about ¼ of full SI flow at this operating point. "A" MU pump was still operating in throttled condition.
t ≈ 5 min.	Operator initiates letdown flow in excess of 140 gpm in additional attempt to control pressurizing level. About 2 minutes later letdown flow is throttled back to about 70 gpm. At this point and continuing for about the next two hours (until E-M relief valve is shut) the amount of primary coolant being lost due to letdown and release through the open E-M relief valve is well in excess of that being added by one throttled MU pump. Therefore, during this approximate two hour period the voids in the RCS were steadily increasing and eventually led to the uncovering of the core.
t = 7:43 min.	R.B. sump pump "A" automatically started on sump high level, presumably pumping about 140 gpm to the miscellaneous waste holding tank through normally open containment isolation valves. (These valves isolate on R.B. high pressure at 4 psig which had not yet been reached). This pump was instead lined up to the auxiliary building sump tank which had a blown rupture disk. This tank later overflowed into the auxiliary building sump and backed up and flooded most of the floor drains in the auxiliary building basement.
t = 8:00 min.	Operator discovered very low level indication in both steam generators which would indicate they were dry. He then verified emergency feedwater system status and found both block valves closed. (The position indication for one of these valves may have been obscured by a caution tag from another valve controller). Operator opened the valves and fed both S/G with relatively cold feedwater causing additional shrinkage of the RCS without sufficient makeup.

t = 10:00 min.	Pressurizer level came back on scale but remained high.
t = 10:19	R.B. sump pump "B" automatically started increasing total pumping rate to about 280 gpm.
t = 10:24 min.	"A" MU pump tripped. Both pumps off for about 16 sec. "A" restarted at 10:40 min.
t = 14:50 min.	RC drain tank rupture disk burst at about 190 psig.
t = 20 - 74 min.	RCS stabilized near saturated conditions at about 1015 psig and 550° F. Operator periodically requested printout of E-M relief valve outlet temp. Reading was not conclusive that discharge was still occurring. RC flow gradually decreased during this period and various RCP related alarms occurred. Various building exhaust monitors showed small increase during this period. Chart recorder for source range instrumentation showed steadily increasing valves during this period. This was indicative of slowly decreasing moderator density in the core but was not identified by operator.
t = 25 min.	High radiation alarm on Intermediate closed cooling system. This monitor is physically located next to R.B. sump and was normally received following a reactor trip.
t = 38 min.	R.B. sump pumps turned off by operator. Since discharge line was still not isolated (This did not occur until 4 psig was reached at about t = 4 hours) it is suspected that R.B. sump water continued to be transported at a low flowrate to the auxiliary building sump due to elevation differences and higher R.B. pressure.
t = 1:14 hour	Operator tripped RCP's in "B" loop due to vibration alarms and fact that pumps had been below allowable limits for 4 pump operation. "B" loop closed to maintain pressurizer spray capability which comes from "A" loop.
t = 1:27 hour	Operator isolated "B" steam generator. It was believed at this point that high R.B. pressure was due to steam leak from "B" steam generator since it was significantly lower in pressure than "A". Lower pressure was probably due to void which had formed in the "B" hot leg and was preventing flow through this steam generator.
t = 1:30 hour	RCS sample indicated 400-500 ppm boron and 4 uc/ml. This was about a factor of ten increase in activity and a factor of two decrease in boron.

t = 1:40 hour	Operators decided to attempt natural circulation on "A" steam generator due to excessive vibrations on loop "A" RCP's. In preparation for this, level in "A" S/G was raised and both "A" loop RCP's were tripped. In subsequent interviews, the operators did not believe they had established natural circulation. However, the increase in source and intermediate range nuclear instrumentation was thought to be due to the boron dilution that measurements had been indicating. In fact, the operator had started an emergency boration cycle prior to this evolution. At about this time, the operator reported that they increased high pressure injection flow. The RCS pressure showed an increase and the source range monitors (SRM) showed a significant decrease which indicated the core voids had collapsed. The operators apparently did not note the significance of this. A short while later the SRM showed an increase of about one decay which again indicated the core was becoming uncovered. The operator again reported that the "emergency borated." This condition remained for about 1 hour and 15 minutes, until after the E-M relief block valve was closed and pressure was increased above saturation.
t = 1:54 hour	RCS hot and cold leg temperature begin to diverge widely. The hot leg temperature went offscale at 620°F in about 14 minutes. The cold leg temperature dropped to about 150°F (apparently due to HPI water).
t = 2:22 hour	E-M relief block valve isolated by operator. Higher temperature readings on this valve finally led operators to believe that it was leaking. This action terminated the small loss of reactor coolant accident and RCS pressure began increasing from it low point of about 1300 psig.
t $\approx$ 2:40 hour	Area radiation monitors alarmed at the sample station and letdown line radiation monitor increased by about a factor of 100.
t = 2:45 hour	Operator opened isolation valves on "B" steam generator in preparation for attempting to restart RCP's. Several attempts were made to start RCP's in "A" loop. Finally a few minutes later RCP-2B was started. It remained in operation for about 19 minutes when it was tripped due to vibrations and a low operating current.
t $\approx$ 2:50 hour	A site emergency was declared. First notice to offsite agencies was initiated.

t = 2:56 hour	Radiation alarm received from condenser off gas monitor. Operator again isolated "B" steam generator due to suspected primary to secondary leakage. A small amount of leakage within allowable technical specifications was occurring in this steam generator prior to the transient. It was suspected that cold feedwater into a dry steam generator may have increased the severity of this leakage.
t = 3:00 hour	RCS pressure increased to 2130 psig. SRM again began to increase slightly.
t = 3:12 hour	Operator opened E-M relief block valve in an attempt to establish normal pressurizer level. SRM count rate decreased to normal indication flow may have started through reactor core. As RCS pressure continued to decrease, SRM count rate again began to increase indicating steam again forming in the core.
t = 3:21 hour	High pressure injection again initiated on decreasing pressure. Both MU pumps are started with full flow. SRM count rate rapidly decreased to normal indicating core void had again collapsed. SR and IR monitors would remain near normal levels from that point on. This is a good indication the core was not uncovered following this restoration of full HPI flow.
t ≈ 3:24 hour	General emergency declared as a result of high radiation levels in the reactor building.
t ≈ 3:30 - 3:55 hour	Operator attempts to control pressurizer level by cycling E-M relief block valve and by stopping "C" MU pump.
t = 3:56 hour	Reactor building isolated by SFAS actuation at 4 psig.
t ≈ 4:00 - 5:15 hour	With pressurizer level restored to about 380 inches and RCS pressure at 1500 psig additional attempts were made to start a RCP during this period. The containment dome monitor steadily increased from about 200 R/hr to about 6000 R/hr. Offsite monitoring teams were dispatched and reported < 1 mr/hr offsite.
t = 4:17 hour	Both operating MU pumps ("A" & "C") tripped. Operator attempted to start one about one minute later. MU pump "B" started about 4½ minutes later. About 10 minutes later MU pump "C" started by operator. Two MU pumps were running at full flow for about the next five hours.

t = 4:38 hour	Steam dump to atmosphere began on "A" steam generator.
t = 5:15 - 7:30	Operator closed E-M relief block valve in an attempt to raise pressure and collapse steam bubbles that they believed were in the loops. Pressure was controlled at about 2000-2200 psig. by cycling E-M relief block valve. Decay heat was being removed mainly by feed (HPI) and bleed (EMRV) process and somewhat dumping steam from "A" steam generator through atmosphere dump.
t = 6:14	RCS activity reported to be 140 uc/cc gross β^{-} γ.
t = 7:30 - 10:30	Operator reduced RCS pressure by opening E-M relief block valve. This was done to insure that the core was covered since at about 600 psig. the core flood tanks would inject directly into reactor vessel on top of core. Once it was assured that the core was covered, an attempt would be made to further depressurize and initiate decay heat removal (the normal long term cooling mode using forced recirculation through an external cooling system) at 400 psig. About an hour later when the initiation pressure of the core flood tanks was reached, indications were that very little water was injected, therefore the operator felt confident that the core was covered. However, the RCS pressure could not be reduced below about 450 psig. which the operators attributed to reaching the saturation pressure of the loops. Decay heat was being removed mostly by feed (HPI and core flood) and bleed (EMRV and pressurizer vent) and somewhat by atmospherically dumping steam from "A" steam generator.
t = 8:30	Steam dump to atmosphere from "A" steam generator stopped at request of corporate management in response to concerns expressed by state government.
t = 9:50	ESF actuation on high R.B. pressure. (Building pressure experienced a short spike to 28 psig. which cleared within 11 seconds) R.B. spray was initiated and was shut off by operator after about 6 minutes. Since this occurred simultaneously with the operator opening the E-M relief block valve, it was believed that noise or an electrical cross connection had yielded a false signal. Some people in the control room reported hearing a dull thud at about this time. This indication is what was later believed to be a hydrogen explosion in containment. Since it caused no evidence of instrument or equipment failure, its significance is questionnable except for indicating the extent of metal water reaction. If it was a hydrogen explosion, it was a localized occurrence based on its duration and effect.

t = 10:30 - 13:30 hour	With RCS at about 500 psig. "A" loop Th decreased indicating that the bubble in the loop had collapsed. This was followed by an increase in Tc which was indication that some natural circulation was occurring. This is thought to be primarily the result of HP injection which was primarily directed to the "A" loop. It was still planned to try to further reduce pressure and go to low pressure injection followed by normal decay heat removal. Decay heat was now primarily being removed by the ongoing feed and bleed process.
t = 13:05	Started to draw a condenser vacuum. Started steaming "A" steam generator to condenser about 15 minutes later.
t = 13:30 - 15:30	Since RCS pressure could not be reduced below about 450 psig. operators decided to repressurize RCS in an attempt to further collapse voids and start a RCP. With E-M relief block valve closed and MU flow at about 500 gpm with two pumps throttled, RCS was increased to about 2250 psig. in about one hour. In preparation for starting a RCP, MU flow was balanced with letdown and an attempt was made to draw a bubble in the pressurizer. Decay heat was now primarily being removed through some natural circulation in "A" steam generator which was steaming to the condenser.
t = 15:33	RCP-1A started for about 10 seconds as per the procedure for restart following loop filling. RCS pressure dropped to about 1450 psig.
t = 15:50	Operator started RCP-1A to establish forced circulation through the "A" loop. RCS pressure dropped from about 2250 to 1380 psig. and eventually stabilized at 1000 psig. Tave dropped to about 290° F and eventually stabilized at about 250° F.