REPORT ON A MISSION TO THE STATES BY MESSRS. B. ROCHE AND A. CAYOL FROM APRIL 1st TO APRIL 6, 1979 ON THE ACCIDENT OF THE THREE MILE ISLAND NUCLEAR PLANT

This report was drafted following a mission to the States from April 1st to April 6, 1979, to collect the maximum data available on the accident which occurred on March 28, 1979 at about 4 a.m. on Unit 2 of the Three-Mile Island nuclear plant near Harrisburg in Pennsylvania. This mission was in addition to the permanent team in Washington which is headed by Mr. P. Zaleski, nuclear attaché at the French Embassy.

It is a synthesis of data which were given by the Nuclear Regulatory Commission (NRC) within the existing agreement between this organization and the Industry Ministry.

It seems already possible to determine and comment the significant events which happened during the first hours of the accident. However, all details concerning the causes and the development of the accident will only be available when the inquiry in process in the United States will be over within 3 to 6 months.

It all started with an incident which generally has no serious consequences and which happened in the non nuclear part of the installation: loss in flow of feedwater to the steam generators. However, following the violation of a technical specification (closing of valves which should have been open), the auxiliary

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feedwater system of the steam generators did not play its role. This resulted in a drying of the steam generators which prevented a good cooling of the core. Parallel to or consecutively, a relief valve of the primary system was not totally closed again, which created conditions for a leak of primary coolant water in the containment.

At this point of the development of the accident which had no important consequences for the plant and the environment, the operator, taking into account an indication of the pressurizer level which was not reflecting the real situation, stopped for several minutes the flow of the emergency cooling system which had automatically started. This action resulted in a lack of cooling for the core which consequences cannot yet be evaluated.

In a second phase which lasted approximately from 5 a.m. to 8 p.m. on March 28, the operator tried to ensure the cooling of the core. First, he took an initiative which rapidly proved to be inadapted to the situation by turning off the main coolant pumps. He certainly feared their damaging due to the effect of cavitation.This stopping of the primary pumps seriously weakened the cooling of the core and resulted in important damage to the fuel bundles (it is estimated that 10 to 25% of them were broken) as well as in a zircaloy-water chemical reaction, causing the release of hydrogen responsible for the formation of a bubble in the upper part of the vessel. However, contrary to some information, there was no noticeable fusion of the uranium oxyde.

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In the evening of March 28, primary pumps were again in operation thus allowing for a satisfactory cooling of the core of the reactor..

As of April 6, 1979, the cooling down of the plant was still a delicate operation as for this purpose it is necessary to use a circuit which is located outside the containement building and, in case of leaks, there may be radioactive releases in the environment.

Radioactive releases in the environment were from two main origins:

- The venting of the primary coolant which run out accidentally in the auxiliary building when the tanks for liquid effluents overflowed. This primary coolant was pumped from the reactor containment building after the beginning of the accident. When the security injection circuit started, it did not trigger the isolation of the containement (a mechanism special to Three Mile Island).
- Voluntary discharges of radioactive gases outside the circuit for processing gaseous effluents which storage capacity run the risk of being saturated following an unusual production of gas caused by the accident.

Although the consequences of the accident on the installation are particularly serious -it is not possible at this time to make a decision on an eventual re-activation of the reactor - radioactive releases in the environment proved to be rather weak. The maximum dose received by the person assumed to be most exposed (who is supposed to remain permanently at the limit of the perimeter of the utility) was below 85 mrem over a period of about five days. This dose is

equivalent to the dose received annually by an inhabitant of the region through natural radioactivity.

In spite of certain difficulties due to coordination problems, the assistance plan was carried out well by local authorities.

Finally, NRC is making a very important effort to analyse the accident. The conclusions will be published in the United States and the technical aspects of this analysis will be held at the disposal of foreign safety organisms within a few weeks.

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MINISTRY OF INDUSTRY

Service Central for the Safety
of Nuclear Installations

Paris, April 20, 1979

JUL 16 1979

Subject: Report of the French experts
mission on the Three Mile Island 2
incident

Dear Sir:

First I would like to thank you for the welcome by NRC to the French experts, Messrs. Cayol and Roche, during their recent mission to the United States. The readiness, they could get - thanks to your intervention - to gather very useful information to accomplish their task in the best way.

I am sending you a copy of the two experts' report with attachments, to let you know how they interpreted the information received by American experts on the Three Mile Island incident.

To inform with utmost clearness the French opinion particularly on the consequences of the incident toward the population and the environment, the authorities will, in short term, publish the report.

Also I will be very grateful, if you would be good enough to let me know NRC staff review, your significant comments in a short time.

In that case, I would be willing to call you by telephone, during next week, before the report is made public.

Thanking you again for your cooperation, best regards,

The Chief of the Service Central

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M. CAYOL

CENTRAL OFFICE OF NUCLEAR FACILITY SAFETY

NUCLEAR PROTECTION AND SAFETY INSTITUTE - IPSN
(NUCLEAR SAFETY DEPARTMENT - DSN)

JUL 12 1979

REPORT

on the

MISSION TO THE UNITED STATES

by Messrs.

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I - REPORT

ACCIDENT AT THREE MILE ISLAND UNIT 2

ON 28 MARCH 1979,

AT APPROX. 4 A.M. (LOCAL TIME)

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INTRODUCTION

This report pertains to a mission organized following the announcement of an accident which befell the Three Mile Island 2 reactor in the morning of 28 March. Its basic objective is to present a preliminary summary of information gathered on the site of the accident, particularly from the Nuclear Regulatory Commission (NRC), the federal agency responsible for nuclear safety in the United States.

Although NPC, in the first few days following the accident, made information available to the public as soon as its reliability could be established, it nevertheless made it clear that its inquiry could not produce conclusive results until weeks if not months later. Consequently, this report cannot provide comprehensive information on all aspects of the accident.

All documents which were obtained in the United States and used in the preparation of this report are included as appendices hereto.

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PART I

GENERAL DESCRIPTION OF THE MISSION

I-1 PURPOSE OF THE MISSION

The mission was organized with the purpose of obtaining the greatest possible quantity of information on the accident at Three Mile Island 2 and deriving the first lessons from it. It was made easier by the existence of a bilateral information exchange agreement signed in 1974 by and between the NRC and the French Ministry of Industry. In addition, the mission filled out the permanent team stationed in Washington under the leadership of Mr. P. Zaleski, nuclear attaché at the French Embassy.

I-2 DURATION OF THE MISSION AND PERSONS CONTACTED

* Mr. Roche arrived in Washington at approximately 12 noon (local time) on 1 April; Mr. Cayol arrived at about 6 p.m. (local time) on the same day.

* Mr. Cayol left the United States in the late afternoon of Thursday, 5 April; Mr. Roche departed the country in the late afternoon of the following day, Friday, 6 April.

We contacted Dr. Jo LaFleur, Assistant Director of the Office of International Programs (OIP) of the NRC, who furnished to us all information bulletins issued by the NRC. The OIP also set up an information meeting for visitors and foreign nuclear attachés on 3 April, as well as the tour of the site on 5 April, in which we were able to have a brief meeting with the plant operators.

We were also present at a public meeting of the NRC Executive Committee held on 4 April. The meeting was intended to determine which specific actions (including possible shutdown) should be taken in regard to operating nuclear plants designed by Babcock and Wilcox (Three Mile Island 1, Arkansas 1, Crystal River 3, Davis Besse 1, Oconee 1, 2 and 3, Rancho Seco). We should recall that

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the Governor of California called for the closing of the state's Rancho Seco facility shortly after the accident at Three Mile Island. The meeting was called partly in response to the Governor's request.

On 4 April we attended a meeting of the Advisory Committee on Reactor Safeguards (ACRS)--the equivalent of our "standing groups"--, which was called to study the accident.

Mr. Zaleski arranged an interview for us with Messrs. Larowski, Frailey and D. Okrent (the chairman, the secretary and a member of the ACRS, respectively) on 4 April.

Finally, we had a telephone interview with an engineer (Mr. ~~Kaeso~~ Caso) from Westinghouse (Pittsburgh) in the morning of 3 April.

I-3 MAIN DOCUMENTS

The principal documents we were able to obtain were as follows:

* The bulletins issued by the NRC for distribution to U.S. leaders and to officials of the NRC itself and of other, foreign organizations. These bulletins, numbered PNO 79-67 through 79-67K, are annexed as Appendix A. They were distributed to all the foreign experts.

* The accident scenario prepared by Babcock and Wilcox engineers. This document, which was kindly provided by Mr. Frailey, is annexed hereto as Appendix B.

* The measures taken by the NRC (inspections and analysis complements) in regard to Babcock and Wilcox reactors presently in operation (Appendix C).

* The report of the above-mentioned meeting of the NRC Executive Committee on 4 April (Appendix D).

I-4 ACTIONS TAKEN

By means of telephone conversations and telex messages, we kept the Central Office of Nuclear Facility Safety and the Nuclear Protection and Safety Institute advised of the latest daily information as it became available. (The

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complete set of telex messages sent is annexed as Appendix E.)

I-5 FOREIGN EXPERTS PRESENT

We noted the presence of engineers from the following countries: The Netherlands, Spain, Belgium, Italy, Federal Republic of Germany, Japan and Mexico. In addition, representatives of the IAEA and the Committee of European Communities were also present.

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PART II

BRIEF DESCRIPTION OF THE FACILITY

Appendix F contains extracts from the plant safety report, which provide general information on the facility as a whole and include diagrams of the primary loop and its relation to the containment building.

It might also be mentioned that Three Mile Island Unit 2 had been put in service at the end of 1978 and that its fuel elements had been in operation for a total of only 60 day equivalents at full power at the time of the accident.

The Three Mile Island plant is operated by Metropolitan Edison, which together with two other local companies has formed a group known as the General Public Utilities Corporation (GPU).

The operation of the two units on the site employs about 320 people.

The principal characteristics of Unit 2 are as follows:

Thermal Power	2,770 MW
Electrical Power (Net)	880 MW

* Fuel Elements

  - UO_2 at 2.5-3% enrichment
  - 177 assemblies, 15 x 15 each.

* Reactor Vessel

Outside Diameter	4.7 m
Total Height	12.20 m
Thickness in current portion	0.21 m

* Industrial Architect: Burns and Roe Inc.

* Turbine Manufacturer: Westinghouse.

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PART III

DESCRIPTION OF THE ACCIDENT

III-1 DESCRIPTION OF THE FIRST SIXTEEN HOURS OF THE ACCIDENT ON 28 MARCH

At a public meeting on 4 April, minutes of which are included in Appendix D, NRC engineers presented to the Commission's directors the accident scenario they had reconstructed from data recorded at the power plant. Only data for the first 16 hours were examined.

Taking as the time origin the moment when the accident started on 28 March (4 a.m. local time), the scenario is as follows:

Time	
0	• Loss of normal supply to the steam generators, for a still unknown reason. This entails automatic shutoff of the turbine.
3-6 sec	• Opening of the pressurizer discharge valve in response to signal of excess primary pressure (2255 psi - 153 bar).
From 9 to 12 sec	• Emergency shutdown of reactor in response to excess primary pressure signal (2355 psi - 161 bar).
From 12 to 15 sec	o Primary pressure falls to 2205 psi (152 bar) and then to 2147 psi (148 bar). The temperature of the hot branch at the core outlet reaches 611°F (320°C). The pressurizer exhaust valve should have closed again at 2205 psi, <u>but it fails to close</u>.
30 sec	• The three steam generator relief supply pumps operate at full pressure. There is no flow through them: <u>the valves downstream of them are closed, in flagrant contradiction of the technical specifications</u>.
60 sec	• The pressurizer level rises rapidly -- "Low" level of steam generators A and B.
2 min	o Automatic cut-in of the high pressure safety injection in response to low primary pressure signal (1600 psi - 110 bar).

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From 4 to 11 min

- The pressurizer level exceeds the dial range. The operator stops the two safety injection pumps at 4.5 min and 10.5 min respectively.
- At 6 min, steam flashing in the core. Primary pressure: 1350 psi (93 bar); core outlet temperature: 584°F (308°C).
- At 7.5 min, the sump pumps come on automatically.
- At 8 min, the operator opens the steam generators' emergency feed valves.
- At 8 min 18 sec, the pressure reaches a minimum in steam generator B.
- At 8 min 21 sec, the pressure (secondary loop side) within steam generator A climbs.

From 11 to 12 min

- The pressurizer level can be read again.
- High pressure emergency core cooling system pumps are turned on manually.

15 min

- Rupture of the explosion membranes of the pressurizer discharge balloon at approximately 200 psi (13.5 bar).

From 20 to 60 min

- Pressure and temperature of primary loop remain stable; primary pressure: 1050 psi (72 bar); core temperature: 550°F (287°C) (boiling conditions).

1 hr 15 min

- The operator stops the two primary pumps of loop B.

"Some damage had certainly already occurred to the fuel rods at this stage, but the worst was yet to come" (NRC citation).

1 hr 40 min

- The operator shuts off the two primary pumps of loop A.

1 hr 45 min to 2 hr

- The core begins to heat up rapidly.
- The core exit temperature exceeds 620°F (327°C) and goes beyond the readout range for 14 minutes.
- The core inlet temperature falls to 150°F (65°C).

From 2 hr 15 min to 3 hr

- The primary pressure rises from 700 to 2150 psi (48 to 148 bar).

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- At 2 hr 20 min, isolation of steam generator B. Discharge of secondary steam into atmosphere through controlled discharge valves.
- At 3 hr, opening of the pressurizer discharge valve (manually).
- At 3 hr, detection of radioactivity in the sump drain system.

3 hr 15 min
- Peak pressure: 5 psi (0.34 bar) in the pressurizer discharge balloon.

3 hr 50 min
- New pressure peak: 11 psi (0.76 bar) in the pressurizer discharge balloon.
- The primary pressure is estimated as 1750 psi (120 bar).
- The pressure in the containment building reaches 4.5 psi (0.31 bar).
- Automatic isolation of the containment building as pressure drops to 4 psi (0.3 bar).

From 5 to 6 hr
- Primary pressure rises from 1250 to 2100 psi (85 to 145 bar).

7 hr 30 min
- Pressurizer discharge valve opened in order to reduce primary pressure.

Note: The discharge valve was operated numerous times. It seems to have remained useable to some extent throughout the accident.

From 8 to 9 hr
- Primary pressure decreases to 500 psi (35 bar).
- The temperatures measured by the core thermocouples (52 measurements) vary widely. Some thermocouples go off the reading scale (620 to 700°F - 325 to 370°C). Boiling is certain to be taking place in part of the core.
- The accumulators are partially emptied into the dome (accumulator load pressure: 600 psi - 41 bar).

10 hr
- 28-psi (1.9-bar) pressure peak in the containment building (probable hydrogen explosion).

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- Start of the containment building sprinkling system, which is then cut off (manually?) after discharging 5000 gallons (20 m^3) of borated water in the building.

13 hr 30 min

- Pressurizer discharge valve closed to increase the primary pressure in order to:
  - reduce the size of a bubble of incondensable materials and/or steam, and
  - make it possible to start the primary pumps.
- Primary pumps of loop A are turned on.
- At this time, the large temperature difference between the core inlet and exhaust (cold and hot branches) indicates that the flow in the core is practically nil.

13 hr 30 min to 16 hr

- Primary pressure increases from 650 psi to 2300 psi (44.8 bar to 158 bar).

16 hr

- Core outlet temperature decreases to 560°F (293°C).
- Core inlet tmeperature increases to 400°F (204°C).
- Steam generator A functions and discharge its steam into the condenser, where a vacuum has been established.

Sixteen hours after the accident, the general condition of the facility is as follows:

* Severe damage to fuel rods. NRC engineers estimate that approximately 10% to 25% of the fuel rods have broken. Their deformation hinders the cooling of the rods in the central area of the core. Nevertheless, later measurements of the primary water radioactivity level show a low concentration of solid fission products, indicating that no significant fusion of the fuel has taken place.

* Existence of a bubble of incondensable vapor at the top of the core. The presence and size of this bubble could be estimated by correlating the pressure variations in the primary loop, the hot and cold branch temperatures and the pressurizer level.

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The bubble consists basically of hydrogen generated by a zircalloy-water reaction which occurred during the core drying stages, when the temperature of the sheaths made such a reaction possible (a value of 3500°F or 1920°C was mentioned).

* Local boiling in the core, as indicated by the core thermocouples.

* Highly contaminated water in the core, with a radioactivity level of 800,000 Ci/m^3 (see Part IV).

* Contamination of the nuclear auxiliary building after approximately 40 m^3 of contaminated water is dumped in it, having been drawn out of the containment building by the pumps. The spillage is due to the overflowing of the liquid effluent treatment tanks.

* Cooling of the core is ensured by the flow of primary water (use of a primary pump) in loop A and removal of the heat through steam generator A. This procedure requires the operation of the make-up water system and, more specifically, a discharge of highly contaminated primary water at the rate of approximately 80 liters/minute toward the chemical and volume control system loops (RCV).

* Let us recall that the primary pressure is on the order of 2000 psi (138 bar) and the hot branch temperature is 560°F (293°C); the core thermocouples indicate the presence of areas of boiling within the core.

III-2 DEVELOPMENTS AT THE FACILITY FROM 28 MARCH TO THE EVENING OF 6 APRIL 1979

We were unable to obtain precise information on the exact evolution of the plant's parameters after the first 16 hours. Nevertheless, the main actions of the plant operator, undertaken under the control of NRC engineers with the concurrence of Babcock and Wilcox engineers, were intended:

(a) <u>With respect to cooling the core</u>, to reduce the occurrence of

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boiling in the core while gradually lowering the primary pressure.

The cold shutdown of the primary loop and its cooling by the shutdown cooling loop were made very delicate for several reasons:

* The presence of a bubble of incondensable gases above the core: the reduction of the pressure which would have required passage over the shutdown cooling loop might have resulted in an increased bubble size and a consequent partial pumping out of the core, with a risk of cavitation in the pumps (or of a potential stoppage of the primary loop flow).

* The pumps and exchangers of the shutdown cooling loop are located in the nuclear auxiliary building rather than in the confinement building, so that there was a risk of radioactive discharges into the environment (possible losses in the loop).

* The deformation of the fuel elements could lead, at low pressure, to the poor cooling of the assemblies.

(b) With respect to the containment building, to reduce the hydrogen content by activating recombiners. This use of recombiners required the use of some 400 tons of lead briquettes to limit the dosage received by the personnel. The lead was not available at the site and was obtained from manufacturers over the course of two days. Some of it was even brought in by air.

The recombiners were put in operation on 3 April, in the following manner: one in service, treating about 1.7 m^3/min, and the other in reserve.

Note: (1) The hydrogen level in the containment building had reached 2.2% just before the recombiner was put into operation, compared to a maximum concentration of 1.7% in the morning of 3 April.

(2) The NRC experts estimated that some 11 days would be required to lower the hydrogen concentration to 1%.

(c) With respect to waste materials, to reduce the quantity of emis-

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sions to the outside (rare gases) by venting the effluent tank degassing products into the interior of the containment building through a temporary line. We were unable to learn when this line was installed.

III-5 STATUS OF THE FACILITY ON 5 APRIL 1979 AND POTENTIAL PROBLEMS

On 4 April, measurements within the core indicated that there were no further traces of boiling inside the core.

* Maximum temperature inside core: 477°F (248°C) (only three thermocouples indicate over 400°F, or 205°C). (The water saturation temperature at 69 bar is 285°C.)

* Primary pressure: 1000 psi (69 bar).

* Average core inlet and outlet temperatures: 280°F (138°C).

The residual power output of the core was of the order of 5 MW. Estimates suggested that the bubble of incondensable gases had decreased significantly. We were not given any quantitative indication of its volume; in comparison, it had amounted to approximately 50 cubic feet (1.4 m^3) on 3 April, and the most pessimistic earlier estimates were in the range of 900 cubic feet at 875 psi (25 m^3 at 60 bar).

* * *

When we visited the site, a Metropolitan Edison representative told us that the company intended to return the plant to natural flow, completely refilling the pressurizer. This operation would eliminate the following problems:

* Availability of the primary pumps--which have undoubtedly been damaged in the accident, so that their permanent operation may be randomly affected.

* Availability of external electricity--a loss of grid power would mean the stoppage of unrelieved pumps.

* Availability of pressurizer level measurements--note that one of

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the three level sensors broke down on 3 April.

Nevertheless, going into natural flow presupposed that establishment of the thermosiphon could be assured. The efforts of Babcock and Wilcox's engineers had this objective in mind, and they even considered conducting a thermosiphon establishment test on another reactor of the same type.

Switching to cooling by means of the shutdown cooling system was not considered in the short run, primarily because of the potential radioactive waste discharge and contamination problems that might result from leakages from the loop outside the containment building, in which a particularly radioactive fluid would be carried.

III-4 HIGHLIGHTS OF NRC'S CONCLUSIONS

In the conclusion of its meeting of 4 April, the NRC indicated that the accident had resulted from the following six causes:

(1) The failure of the steam generator (ASG) relief feedwater system, related to the closed position of the isolation valves of the system, which violated the technical specifications. The valves had no doubt been closed two weeks earlier in order to permit maintenance or test operations in the loop.

(2) The failure of the pressurizer discharge valve to close completely after the primary pressure drop.

(3) The unusable indications given by the pressurizer water level.

(4) It was not remembered that the containment building is automatically isolated by means of the emergency core cooling system in the Three Mile Island reactor. Nevertheless, the activation of the isolation is supposed to have stopped the water removal pumps, thus preventing the overflow of the primary effluent tanks and consequently limiting the level of waste vented to the outside.

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(5) The operator prematurely turned off the emergency core cooling system.

(6) The shutdown of the primary pumps contributed significantly to worsening the damage to the fuel rods.

III-5 DISCUSSION

The accident may be subdivided into successive stages.

Stage 1 - from t = 0 to t = 60 minutes, approximately.

In this stage, an accident occurs, resulting from the combination of two events:

(a) The loss of the normal steam generator feedwater system--a classical transient which has been thoroughly considered in safety studies and which entails no damage to the reactor.

(b) The failure of the steam generator relief feedwater system, linked with a violation of procedures.

This combination of events had not been considered in the facility's safety studies.

In the reactor, there was a significant imbalance between the heat produced in the core (nuclear reaction during the first 10 to 15 seconds, followed by residual heat after the control rods dropped down after this period) and the power removed by the steam generators, in which the water level fell rapidly because of the lack of feedwater.

Note: The forced circulation Babcock and Wilcox steam generator contains only a small amount of water; the rate of flow through it is five to ten times faster than in a U-tube recirculating steam generator (Westinghouse, Combustion Engineering and KWU).

The behavior of the plant at the beginning of an accident of this kind is characterized by a rapid heating of the primary fluid accompanied by a simi-

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larly rapid increase in pressure. The pressurizer discharge devices open normally. (According to the NRC, the two safety valves of the pressurizer did work properly.) The failure of the discharge valve to close can be explained, since it was forced to operate in water or in a water-steam mixture, conditions for which it was not designed.

The failure of the discharge valve to close undoubtedly aggravated the event, inasmuch as it caused a depressurization of the primary loop and a concurrent deterioration of the heat exchange at the fuel rod sheaths.

The increase in the measured pressurizer level is related to the expansion resulting from the depressurization. It should be recalled that the pressurizer level is evaluated on the basis of the pressure difference in a water column, which measure becomes less significant in the case of a water-steam emulsion.

The closing of the emergency cooling system by the operator between t = 4.5 minutes and t = 12 minutes led to a mass boiling in the core, which may have resulted in damage to the fuel rods (deformation, and possible ruptures). The reasons for the closing may be related to the operator's wish to keep the pressurizer level within the instrument scale.

Several incidental or induced events took place during this stage:

(a) The automatic start-up of the containment sump ~~exhaust~~ pumps, resulting from the presence of water at the bottom of the sump. The water undoubtedly came partly from the pressurizer discharge ~~balloon~~ tank (total volume, 24 m^3), whose valves are set to approximately 7.5 bar. The exhaust valves were in operation until t = 3 hours, at which time the containment vessel was isolated in response to the "High Containment Pressure" signal.

(b) The rupture of the discharge disks (at about 13 bar) of the pressurizer discharge tank. This rupture was no doubt due to the large quantity

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of water discharged by the discharge valve.

In any case, the opening of the steam generator feedwater isolation valves by the operator at t = 8 minutes, together with the renewed cut-in of the emergency core cooling system at t = 12 minutes, led to the reactor being in a nearly stable condition at t = 60 minutes, the end of the first stage. But conditions in the secondary loop (pressure of the order of 64 bar and temperature of the order of 280°C) made it impossible to ensure a temperature clearly lower than the saturation temperature in the primary loop, where the pressure was 70 bar. This explains the presence of steam in the primary water.

Stage 2 - from t = 1 hour to approximately t = 16 hours.

At t = 1 hour 15 minutes and t = 1 hour 40 minutes, respectively, the operator closes the primary pumps (two per loop) of loop B and loop A.

No reasons were officially made known for this shutoff. According to the NRC's Dudley Thompson, they are probably related to feared or observed cavitation in the pumps. The primary fluid contained significant steam voids which could in fact cause cavitation events.

Thus, it may be supposed that the operator hoped to cool the core by natural flow. Nevertheless, according to NRC experts, the presence of steam "locks" in the primary water (at the U bends of the hot pipes at the top of the steam generators), in conjunction with the condition of the secondary loop, undeniably prevented the thermosiphon from becoming established. The exact reasons for the stopping of the pumps, which now seems to have been a serious error in judgement, will not be known until the completion of the detailed inquiry now being pursued by the NRC.

In any case, the core was quite empty and had no cooling at all after the loop A pumps were turned off at t = 1 hour 40 minutes. The core (or at least its central portion) dried out several times, leading to overheating of

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the sheaths and significant zircalloy-water reactions. As a result, hydrogen was released. It should be noted in this regard that the design of the safeguard systems is intended to ensure that almost no zircalloy-water reaction (a strongly exothermic reaction) occurs in any design accident, by limiting the temperature of the sheaths to values below 1200°C.

It is difficult to discern the exact reasons for the various actions taken by the operator during this stage of the accident. One may suppose, nevertheless, that the operator found himself faced with an unknown situation and:

(a) attempted to determine just what the situation was, and

(b) tried various approaches leading to reestablishing a stable flow in the core.

In particular, one may attempt to explain his various actions as follows:

* Handling of the Pressurizer Discharge Valve

Preliminary Note: It is not exactly known whether the operator controlled the valve itself or the (normally open) isolation valve located upstream of it.

The operator closed this valve in order to increase the primary pressure with a view to condensing the steam in the loop; the condensation was meant to make the primary fluid homogeneous and thus allow the thermosiphon to function. But the only result of the ~~maneuver~~ action was to increase the primary pressure without condensing the steam in the core.

The operator then tried the opposite action. By opening the pressurizer relief valve, he sought to reduce the primary pressure, thinking he would thus be able to go into the shutdown cooling system (NRC information). The lowered pressure resulted in an increased flow of the high-pressure emergency core cooling system and the automatic start-up of the accumulators (opening pressure, approximately 40 bar), whose operation was not inhibited

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as it should have been under the normal sequence of passage to the cold shutdown mode.

In conclusion, it may be supposed that the operator again closed the pressurizer valves at approximately t = 13 hours 30 minutes, in an attempt to increase the primary pressure in order to condense the steam in the entire loop and thus make it possible to start the primary loop A with limited cavitation. His choice of loop A is explained by the fact that the pressurizer is in this loop.

The latter operation also made it possible to properly cool the core. The reasons for the success of the operation are difficult to determine at this time. The arrival of cold water from the accumulators must have contributed to it but is not sufficient to fully explain it. The condition (pressure) of the secondary loop also must have played a part. We cannot formulate a definitive conclusion without knowing this condition.

* Isolation of the Steam Generators

The reasons for the isolation of the steam generators are not fully known in detail. Perhaps the operator acted in this manner in order to prevent discharges of the contaminated primary fluid into the environment, which might have resulted from preexisting leakages in the steam generator pipes.

At any rate, the operator let out the steam from steam generator A (the only one that was back in operation) through the condenser at t = 16 hours, in order to prevent direct venting of steam into the atmosphere.

* General Remarks

(1) The NRC was advised of the accident about 7-8 a.m. (that is, at t = 3-4 hours); NRC "Inspection and Enforcement" teams arrived at the site at approximately 10:30 a.m. Babcock & Wilcox's engineers were also notified (at the same time as the NRC). Thus, it can be said that by 10-11 a.m. (that is,

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at t = 7-8 hours) the operator was being advised by technicians who undoubtedly were able to suggest courses of action to him.

(2) In his presentation on 3 April, Dodg Thompson of the NRC stated that the operator, acting upon his recommendation, had certainly attempted to prevent damage to the facility; this should be kept in mind when studying the shut-off of the primary pumps at t = 1 hour 40 minutes.

Stage 3 - From t = 16 hours (about 8 p.m. on 28 March) to 5 April.

* Core Cooling by Means of Forced Circulation Through Loop A

The information obtained from the NRC indicates that the operations undertaken during this stage were intended to reduce the primary pressure with a view to an eventual cold shutdown. It may be said that the primary concern of the operator was to reduce the residual power and that the teams of experts under the leadership of Babcock & Wilcox tried to estimate the size of the gas bubble by looking at the variation in the pressurizer level as the temperature and pressure changed.

The remarks that might be made in regard to this stage of the accident have to do with how the size of the gas bubble was estimated by the experts and with the actions taken to reduce it.

* Formation of the Bubble

The hydrogen content in the reactor vessel, which was higher than 2%, clearly indicated that a significant zircalloy-water reaction had taken place during stage 2. This reaction was the cause of the formation of the gas bubble above the core; the gases contained in the fuel rods damaged in the accident contributed only slightly to the formation of the bubble, according to the NRC.

* Estimation of Bubble Size

The NRC (or the operating company) requested a working group consisting of technicians from various organizations (NRC, builders, universities)

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to develop a correlation that would enable the volume of the bubble to be estimated from available parameters (pressure, temperatures and pressurizer level). The group calculated a suitable correlation within two days.

* Reabsorption of the Bubble

According to the NRC, the bubble was reabsorbed because of:

o the transfer of the gases through the hot branch A toward the pressurizer, from which they were removed inside the containment building by means of a remotely controlled opening event;

o the partial recombination of the hydrogen and oxygen;

o hydrogen leakages in the control rod mechanism housings (this explanation is open to doubt, in our opinion).

It is also undeniable that the improved calculation precision made it possible to eliminate the safety margins initially adopted in estimating the volume of the bubble.

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PART IV

RADIOLOGICAL ASPECTS OF THE ACCIDENT

IV-1 SOURCES OF WASTE DISCHARGES

Radioactive discharges into the atmosphere had three sources:

(1) The assumed leakage from steam generator B may have resulted in a release of rare gases and iodine into the environment in the first few hours after the accident. We have been unable to obtain any estimate of the amounts released in this manner.

(2) The overflow of the head tanks of the liquid effluent treatment system resulted in the spillage of approximately 40 cubic meters of highly contaminated water in the auxiliary building. After filtration (absolute filter and iodine filter), the ventilation system of the auxiliary building ejected the rare gases (krypton and xenon) contained in the water.

These uncontrolled emissions were responsible for the radioactivity observed near the facility and in the plume in the first few days following the accident. On a date which was not specified, the company installed vinyl covers over the water contained in the auxiliary building, in order to reduce the rate of degassing and, consequently, the amount of radioactive emissions.

(3) The exposure to air of the primary water, which was removed at the rate of 80 liters/minute by the volume and chemical control system, led to periodic controlled emissions. In fact, the radioactive rare gases initially dissolved in the primary water were collected in tanks designed for storage and treatment of gaseous effluents. The intake of gas was higher than the storage capacity and, in order to prevent untimely emissions through the valves of these tanks (venting pressure, 7-8 bar), the company deemed it more advisable (with the agreement of the NRC) to engage in controlled releases.

At a later, non-specified date, the company returned the gases to the

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containment building, in order to reduce the level of emissions into the environment.

Moreover, it appears that the radioactive discharges into the river resulted from a conscious decision (made with NT authorization): 230 cubic meters of slightly contaminated waste water were released.

IV-2 EXPOSURE OF PERSONNEL

In the morning of 29 March, during a primary water sampling operation in the nuclear auxiliary building, two employees (one operator and one chemist) received total doses of 3.1 and 3.4 rems respectively. The maximum allowable dose for workers is 5 rems per year and 3 rems per three-month period.

Ten other persons in the duty team received doses of the order of 2 to 3 rems in the first few hours after the accident.

Finally, by 5 April the exposure levels in the control room were 0.1 mrem/hr (compared with 0.4 mrem/hr on 2 April). Masks were not required, but the operators did wear them, out of precaution, during a few hours after the accident.

IV-3 PRIMARY WATER RADIOACTIVITY AND DOSIMETRY IN THE BUILDINGS

IV-3.1 Primary Water Radioactivity

Analysis of a sample of primary water indicated a contamination level of 800,000 Ci/m^3 (confirming the significant rate of emission of fission products in the core).

The radioactivity, which amounted to a dosage rate of 1,000 mrem/hr from a 100 cm^3 sample, was due primarily to rare gases (xenon and krypton), iodines, cesium and, to a very small extent (we were unable to obtain the exact results of the isotope analyses), to non-volatile products (strontium). This indicates that no appreciable melting occurred in the fuel elements.

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IV-3.2 Radioactivity and Exposure Levels Inside the Containment Building

No exact values were provided of the radioactivity level inside the containment building. Values of 700 Ci/m^3 on 31 March and 70 Ci/m^3 on 2 April were announced by the NRC and then denied.

The radiation sensors installed within the containment building indicated conflicting values, namely:

* 10 to 20 rad/hr at the service floor,
* 10,000 to 30,000 rad/hr at the top of the building.

We cannot guarantee the accuracy of these readings, which were given to us orally. We were also told that the sensor at the top of the building was a shielded but unsealed sensor, intended to measure the direct radiation from the core, so that the infiltration of radioactive gases under the shielding would be responsible, according to our informer, for the very high reading.

In our opinion, the sensors inside the containment building were not designed for such levels of radioactivity, so that the values they registered had no relation to reality.

On the other hand, radiation measurements in contact with the outside wall of the containment building (approximately 1.2 meters thick) indicated a dosage of less than one mrad/hr. Considering that the containment building walls have an attenuation coefficient of 10^7, the radiation level inside the building certainly did not exceed 10,000 rad/hr.

IV-3.3 Radioactivity Levels Inside the Auxiliary Building

The radioactivity levels inside the nuclear auxiliary building are likely to have caused access problems, including, among others:

* the difficulty of installing the hydrogen recombiner;
* the dosages in contact with the gaseous effluent storage tanks (60 rad/hr), which prevented certain interventions;

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* the dosage rates (10 rad/hr) after the transfer of contaminated water to the auxiliary building.

IV-4 EXPOSURE OF THE PUBLIC

IV-4.1 Direct Exposure

Wind direction on 28 March was toward the north-northeast, so that the town of Middletown (12,000 inhabitants), five miles distant from the reactor, was in the path of wind from the plant.

Data from two of seventeen dosimeters which have been installed in fixed stations for three months show accident-caused 32-hour integrated dosages of:

* 65 mrad, 700 meters north of the reactor, and
* 22 mrad, 1,000 meters north-northeast of the reactor.

These values do not include the radiation due to the rare gas emissions after noon on 29 March.

The occasional uncontrolled releases of rare gases on 29 and 30 March resulted in transient readings of up to 25 to 30 mrad/hr at the facility boundary. They lasted for only a short period (certainly less than one hour).

The NRC's estimates of the public's exposure balance over the five days following the accident are as follows:

* hypothetical maximum individual dose: 80 mrem;
* mean dose for the 2,000 inhabitants nearest the plant: 9 mrem, compared to the annual dose due to background radiation, equal to more than 100 mrem.

By 3 April, the dosage rate outside the facility had fallen to less than 0.01 to 0.04 mrem/hr.

IV-4.2 Iodine Contamination

A total of 152 atmospheric samples were taken. Only eight of them

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indicated iodine-131 activity levels from 0.3 to 2.5 pCi/m^3. The highest observed activity was equivalent to one-fourth of the maximum allowable concentration in the United States for a continuous emission (U.S. regulation: 10 CFR 50, Appendix 1).

Within a 20-kilometer radius of the power plant, a total of 56 milk speciments were obtained (cow's milk and goat milk) in 20 farms. The measurements showed that 19 samples were not interpretable and the remaining 38 were lower than the measurement sensitivity threshold. A detailed analysis of nine samples revealed radioactivity levels ranging from 10 to 40 picocuries of iodine-131 per liter of milk. In comparison, th Federal Department of Health sets a limit of 12,000 picocuries of iodine-131 per liter of milk before cattle are required to be fed on corn feed.

Note: (1) The maximum allowable concentrations in France (Decree of 20 June 1966) are as follows for iodine-131:

* in air: 200 pCi/m^3 in soluble form;
* in water or milk: 1,000 pCi/m^3 in soluble form.

(2) The decrees of 10 August 1976 limit the mean weekly volume-specific activity for a normally operating 3,000-MW reactor unit to a maximum of 0.2 pCi/m^3 for aerosols (essentially iodine).

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PART V

PROBLEMS RELATED TO THE EMERGENCY PLAN

V-1 THE TWO EMERGENCY PLANS

A distinction should be made between:

* The power plant's internal emergency plan, which is the plant operator's responsibility as long as the accident is unlikely to affect areas outside the plant site. In the United States, this plan is regulated by Appendix E of 10 CFR 50 (Code of Federal Regulations, equivalent to France's Journal officiel).

* The external emergency plan, which is the responsibility of the state governor and local authorities. This plan is not covered by any specific regulation. It is covered only by an exchange of letters with the NRC, which grants its approval in principle to the provisions of the plan, and the local authorities, who strive to implement certain means in case of an accident. Indeed, it appears that the plan of the State of Pennsylvania had not been fully developed; none of the American states which have nuclear facilities have an NRC-approved emergency plan at this time.

The emergency plans designed to cover accidents in civilian nuclear facilities seem to be valid as well in the case of accidents in military installations.

V-2 RESPONSIBILITIES OF THE VARIOUS PARTIES

It should be made clear that the decision as to whether or not to evacuate the population is the responsibility of the State Governor. The NRC plays only a consultative role in this regard. In view of this situation, the NRC could not provide to us exact indications on the implementation of the emergency plan.

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Evacuation operations are the responsibility of local authorities (State and Counties).

In regard to the various technical decisions that must be made with respect to the operation of the plant after an accident, the plant operator submits its proposed solutions to the NRC, which should give its concurrence. But the operator is responsible for the consequences of its actions, nevertheless.

V-3 DEVELOPMENT OF THE EMERGENCY OPERATIONS

Although the accident occurred at approximately 4 a.m. on Wednesday, 28 March, the NRC and the State of Pennsylvania were not informed of it until nearly 7 a.m.; this may mean that the plant operating company was not immediately aware of its seriousness. The alarm was given at about 10 a.m.

Starting on the same morning of the 28th, 40 people from the Brookhaven laboratory (New York) arrived at the site with their equipment; this organization is particularly well-trained in obtaining soil and water samples.

Several NRC mobile units arrived at the site in the afternoon of 28 March, charged with the tasks of collecting ~~specimens~~ samples and, if necessary, undertaking decontamination measures. U.S. Army units from the nuclear test center in Nevada arrived with airplanes and helicopters in the afternoon of Thursday, 29 March. These units were responsible only for collecting atmospheric specimens.

On Friday, 30 March, experts from the federal Food and Drug Administration reached the site.

According to information obtained from the NRC, the first few days immediately following the accident seemed to be characterized by some difficulty in ensuring proper organization and coordination among the various ~~intervening~~ parties (local authorities, federal agencies, private corporations). These

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were most apparent in the following respects:

* The complete saturation of telephone lines in the area affected by the accident. The telephone company quickly arrived and very rapidly installed several direct lines to remove the bottleneck.

* The difficulty encountered in using the various specimens collected by the different sampling organizations, due to the difficulty of adequately referencing the collection sites and the lack of sample standardization; three fourths of the milk and water specimens collected in the first two days after the accident were not usable.

On Friday, the operating company decided to permit significant emissions at the stack (1.2 rem/hr, measured at the stack). This led the governor, acting upon a suggestion from the NRC, to go on television to recommend the evacuation of pregnant women and pre-school children within a radius of five miles (8 kilometers) of the power plant. This recommendation was interpreted by the population as an evacuation order. Within a radius of 8 tp 10 miles (13 to 16 kilometers) around the plant, residents attempted to leave their towns, giving rise to numerous bottlenecks and traffic jams. Cars lined up to top off their tanks at gas stations, which were able to meet the demand; and depositors lined up at banks, withdrawing a total of 10 to 15 million dollars in one and a half days.

According to information we were able to obtain, the man in charge of the emergency plan at the local level was not in favor of the evacuation. It seems that a large part of the people who left their homes soon returned. In fact, about 15,000 people (pregnant women and young children) remained away from their homes for a prolonged period.

On Tuesday, 2 April, the authorities completed the study of an evacuation plan covering the 600,000 inhabitants of the area within a radius of 20

miles (32 kilometers) of the power plant. The plan would no doubt have been implemented if it had been necessary to take specific measures to suppress the bubble at the top of the reactor core, or if the hydrogen content inside the containment vessel had reached a dangerous level (above 4%).

Discussions on the effectiveness of ingesting iodine as a preventive measure had been taking place for many months; this non-radioactive iodine is potentially capable of saturating the thyroid and thus preventing the absorption of radioactive iodine. The Three Mile Island accident put an end to these discussions; on Saturday, 31 March, those responsible for the emergency plan concerned themselves with the distribution of potassium iodide doses to the inhabitants. As it happens, however, no potassium iodide was available on U.S. soil in the morning of 31 [sic] March. Thanks to the efforts of an American chemical company, 50,000 doses of the chemical had been brought to the Harrisburg airport by Satusday evening. None was distributed, not even to the power plant's employees, since the company management though it useless.

Furthermore, the authorities recommended that cattle be fed prepared feed, but the sale of milk was not prohibited.

V-4 CONCLUSION

In spite of certain difficulties during the first few hours or even the first two days after the start of the accident, the local authorities were able to adequately manage the implementation of an emergency plan.

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PART VI

SITE VISIT

On Thursday, 5 April 1979, the NRC arranged a visit to the site by car. The tour included members of the NRC, foreign visitors and foreign embassy nuclear attaches (see list in Appendix G).

It was not possible to enter the property of the power plant. We were able, however, to look at the plant from a lookout on the left bank of the river, to the right of the island on which the plant is located. The only sign that there was anything at all wrong at the facility was the presence of journalists and technicians whose mobile units were parked near the lookout.

Some cows grazed on meadows a few kilometers from the power plant. We noticed that a small ornamental plant store halfway from the nuclear plant and the town of Middletown remained open.

Going through Middletown, which lies about six kilometers from the plant, at about 2 o'clock in the afternoon, we noticed no signs of anything unusual--no extra police, a few children and adults in the streets. In short, the town did not appear deserted, although it also did not seem too lively. We could not say whether the many houses in the countryside around the town were occupied or not.

While in Middletown, we found a representative of the power company, who was able to briefly answer a number of questions.

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PART VII

ADDITIONAL INFORMATION OBTAINED

VII-1 ORGANIZATION ESTABLISHED BY THE AUTHORITIES

VII-1.1 NRC Personnel

From the start of the accident, the NRC installed an appropriate organization with the objective of ensuring control of the situation, centralizing and distributing information to the responsible parties, and informing the media. In addition, the NRC made all appropriate arrangements to obtain the greatest possible amount of information in order to permit a detailed analysis of the accident to be made (hearing of all persons who were in the control room at the time of the accident). This analysis will be made public as soon as it is available, within two to six months.

PNO 79-67D (Appendix A) lists the personnel assigned to the site by the NRC, which included:

* 17 inspectors, from three Inspection and Enforcement regions
* 44 ~~radiological protection~~ Health physicists agents from three I&E and National Reactor Regulation (NRR) inspection regions
* 3 ~~spokesmen~~ public affairs officers to provide information, from the NRR
* 19 NRR specialists

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The NRC personnel and at their disposal significant amounts of equipment, including in particular:

* 3 DOE (Department of Energy) helicopters
* 2 DOE laboratory ~~trucks~~ Vans
* 1 NRC instrument ~~truck~~ Vans .

I&E also assembled a crisis staff at its Bethesda facilities, consisting of specialists itself and other NRC divisions. (Although NRC head-

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quarters are located in Washington, D.C., most of its facilities are in Bethesda, Maryland, 20 kilometers from downtown Washington.) This crisis staff centralized the information and published bulletins with the latest developments of the situation (NPO or Preliminary Notification of Event or Unusual Occasion, which are included in Appendix A).

The bulletins were addressed only to the NRC's leaders at the start, but after 30 March they were also distributed to other federal organs, including:

* EPA - Envorinmental Protection Agency
* FDA - Food and Drug Administration
* DOE - Department of Energy,

and in particular to a "White House Situation Room", a crisis organization specially created by the President of the United States.

VII-1.2 Other ~~Organs~~ Agencies

We noted that the personnel of the Middlebourg air base, 5 kilometers from Three Mile Island, had been placed on alert.

We were also told, moreover, that the total number of federal government personnel (including the NRC) mobilized to the site was of the order of 270 persons.

VII-1.3 Press Information

We did not collect exact information on this matter. We were only able to learn, through NRC engineers, that the U.S. President had requested that all information be given to the press through the NRC, on or about 30 March.

The PNO's annexed in Appendix A were not distributed to the press. However, the NRC did make all the information they contain available to the press in its press conferences.

Generally speaking, it was evident that the NRC wished to make in-

formation available to the public within as short a period as possible.

VII-1.4 Information for Foreign Experts

The experts sent by foreign countries were welcomed by the NRC Office of International Programs, which organized the following activities for them, in particular:

* a meeting on 3 April, in which an NRC inspector replied very openly to written and oral questions put to him; and
* a site visit on 5 April.

Furthermore, OIP also distributed all PNO's to the foreign representatives as soon as they were issued. Finally, the NRC informed the foreign representatives of the conclusions of its analyses of the accident.

VII-2 REACTIONS AT THE TIME OF THE MISSION

We have not analyzed the public, press or industrial reactions to the Three Mile Island accident. We were able to observe some significant reactions, however:

* Nuclear Controversy. The accident will give new strength to a controversy that had been losing speed. In particular, it will be an excellent springboard for Ralph Nader, whose public opinion ratings had been in a marked decline. The cartoon published in a local paper (York Daily Record) and enclosed in Appendix H is significant. In conclusion, opponents of nuclear power distributed the tract a sample of which is included in Appendix I at the public meeting on 4 April.

* Press Reactions. By reading the daily papers (The Washington Post, basically) and listening to the radio and TV, we were able to observe that the media kept the public informed of the situation from one hour to the next. The newspaper articles, as well as the radio and TV commentators, were extremely

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measured and could in no way lead to panic reactions. Furthermore, the press started to discuss the financial implications of the accident as early as Tuesday, 3 April, at which time the situation had been brought under control.

VII-3 THE CASE OF OTHER BABCOCK & WILCOX REACTORS

The NRC took very quick action--on 1 and 5 April 1979--in regard to other Babcock & Wilcox reactors under construction or in operation (see Appendix C).

These actions, which are detailed below, pertained essentially to:

* a special inspection effort;

* additional analysis;

* expected changes in facilities and operating procedures.

In any case, the NRC did not deem it necessary to take more rigorous measures with respect to these reactors at its meeting of 4 April, since the results of its inquiries were not yet known.

It should also be noted that copies of the PNO bulletins were distributed to all parties holding reactor construction or operation permits, regardless of the reactor designer.

The questions raised by the NRC in its Bulletin 69-05 Revision A, of 5 April 1979 (see Appendix C), may be summarized as follows:

A - Study of the accident, in order to examine what errors were made, study the usual behavior of the reactor at the time of incidents such as feedwater loss, and make personnel more aware of the seriousness of the consequences of wrong actions.

B - Verify the safeguard systems maintenance and test procedures, and particularly the retention of the functions provided by these systems during tests and their later return to proper operating position.

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CONCLUSION

This conclusion is voluntarily limited to the analysis of the development of the accident, the only subject on which the NRC was able to furnish exact information.

First of all, we must point out that the NRC did not hesitate in making available the greatest possible amount of information to the foreign visitors who came to learn about the accident.

We believe it is premature to attempt to drawn definitive conclusions from the accident, particularly with respect to its exact causes (design defect, defective operation and control procedures, etc.); it will be necessary to wait for the results of the inquiry now being conducted by the NRC.

It may be noted at this time, however, that the accident followed the non-availability of the steam generator emergency feedwater supply system after the violation of a technical specification which called for the valves in this circuit to be kept open.

The consequences of the resulting accident sequence to the facility were not considered in the plant safety studies and operation procedures. The operator thus found himself in an unexpected situation and made inappropriate decisions--shutting off the emergency core cooling system between t = 4.5 minutes and t = 12 minutes and, particularly, shutting off the primary loop pumps at t = 1 hour 40 minutes. The latter operation had the most serious impact on the core.

The failure to close the pressurizer valve seems to us to be more a consequence than a cause of the accident, even though it may have contributed to the core cooling difficulties.

Furthermore, the operator was undoubtedly strongly hindered in his

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appraisal of the situation by the absence of indications of the condition of the primary loop (water level in the pressurizer). It is very likely that, during the first stage of the accident, he was not aware of the exact condition of the facility. In addition, the very rapid drying time of steam generators of the Babcock & Wilcox type allowed him too little time to take more well-considered decisions, at least in the initial stages of the accident. The operator does not seem to us to have known how to take advantage of the possibilities offered him by the secondary loop.

It is important to note that the accident as a whole had severe consequences for the fuel elements, although no ~~significant~~ measurable melting ~~fusion of~~ the fuel took place. There occurred a strong zircalloy-water reaction, which undoubtedly had a greater impact than that of the reaction traditionally contemplated for certain ~~foreseeable~~ Design Basis accidents (such as complete rupture of a large pipe in the primary loop) when conducting the safety studies for power plants of this type. The reaction resulted in the formation of a bubble of incondensable gases which contributed greatly to preventing the operator from bringing the reactor back to a safe condition.

Nevertheless, the safeguard systems normally installed in the ~~facility~~ ~~Design~~ facility to diminish the consequences of accidental sequences different from the one that actually happened were able to play their part in preventing unacceptable emissions into the environment. In this regard, the presence of iodine ~~traps~~ filters in the auxiliary building ventilation system made it possible to avoid unacceptable emissions that might have led to a population evacuation.

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