CONSULATE GENERAL OF THE
UNITED STATES OF AMERICA

Bombay 400 026

June 25, 1979

Mr. Joseph M. Hendrie
Chairman, U.S. Nuclear
Regulatory Commission
Washington, D.C. 20555

Dear Mr. Chairman:

I am enclosing an analysis, prepared by the Indian Department of Atomic Energy (technical addressee in India), of the NRC material regarding the Three Mile Island Accident. As you will note, Indian AEC Chairman Sethna hopes the NRC will give him its views on his analysis of the NRC data on the accident. It will be useful to him in improving Indian nuclear safety measures.

Chairman Sethna has personally told me many times how useful your bulletins on Three Mile Island were to him. They permitted him to assure his Parliamentary Committees that the IAEC was drawing the appropriate lessons from the accident and that such an event was not likely to occur in one of India reactors, which are not PWR's.

I want to add my own thanks to the Commission, as your prompt and comprehensive flow of information did much to reassure the local people that our government was acting responsibly toward its foreign partners during those difficult days.

Sincerely yours,

William F. Courtney

William F. Courtney
Consul General

Enclosure: Sethna Letter Dated June 14, 1979
with Attachments.

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H.N. Sethna,
अध्यक्ष
CHAIRMAN

भारत सरकार
परमाणु ऊर्जा आयोग
GOVERNMENT OF INDIA
ATOMIC ENERGY COMMISSION

छत्रपति शिवाजी महाराज मार्ग
बम्बई-४०० ०३९.
CHHATRAPATI SHIVAJI MAHARAJ MARG
BOMBAY-400 039.

No. 13/4/79-ER

June 14, 1979.

Dear Mr. Courtney,

You have been kind enough to send us various NRC reports on the recent incident at the Three Mile Island Nuclear Power Plant at Harrisburg in the United States. I enclose a brief note containing some of our comments on the chronology of events in this incident which you may consider conveying to the NRC. We would appreciate receiving whatever additional information the NRC may be able to provide us as well as their views on the comments made by us. This would not only help us to arrive at a better understanding of the incident but also put us in a better position to review the safety features of the nuclear power plants in this country.

Yours sincerely,

(H.N. Sethna)

Enc. As above

Mr. William F. Courtney,
Consul General,
Consulate General of the United States of America,
Bombay.

answered 7/20/79

THREE MILE ISLAND REACTOR ACCIDENT

Comments on Chronology of Events

The appendix attached is an indication of our assessment and comments on the Unit 2 Accident. Considerable information is needed for further detailed understanding of how the accident exactly occured and propagated.

1) The presence of moisture in instrument air lines had evidently caused the accidental closure of feedwater circuit valve which failed. How total loss of feedwater was caused by the closure of a single valve, has to be detailed. It is guessed that valve is probably near the polishing unit where the feedwater headers are understood to combine into a single line. It is reported that the turbine trip occured normally but the actual signal causing the turbine trip has a lot of relevance namely i.e., whether it tripped on lower boiler level, lower boiler pressure or any other signal from the feedwater pumps etc.

2) The logging computer print-out indicates the reactor scram at 2355 psi while the relief valve set point was at 2250 psi. It is unlikely that a PWR system is designed to open relief valves before pressure scram. It is known in practice relief valves frequently fail to sit back. Dependance on electromatic relief valve, hence seems to be unreliable. Also clarification is needed whether these relief valves are provided for possible operator intervention to control sudden pressure surges in case the pressurizer cannot cope with the surge.

3) It appears from subsequent bulletins issued by NRC (79-06 A & B) that the operator does not have any direct indication either of the condition of the relief valves or the surge receiving tank, pressure to ascertain whether the relief valve is leaking. This inadequacy has been rectified by the NRC Directives to provide direct indications regarding the above to the operator.

4) Since the logging computer seems to have a three second cycle it is likely that the reactor did scram at the correct pressure but the computer prints out a higher pressure at the time of scram. This point needs clarification. It is presumed that the pressurizer size and the spray system in it have been designed adequately for meeting the transient pressures caused by a turbine trip. If it were so, one would make a guess that this spray did not function adequately or

if there is inadequacy in the design one would have expected that the reactor scram should be initiated directly from a turbine trip. This would have eliminated the initial six full power seconds energy surge into the primary system. This area needs detailed investigation by a complete transient analysis of the power plant.

5) Pressurizer level indication begins to rise rapidly, which caused the operator to trip the HPCI and then subsequently rectify the mistake and restore the HPCI. It is felt that the level surge happened mainly due to swelling of the bulk of the primary fluid due to release of dissolved hydrogen and partial vaporisation along the track of the depressurisation wave front (caused by the stuck relief valve) and not because of flooding by the HPCI or the bulk volume expansion of the primary fluid. Also a point is to be considered whether with a stuck open relief valve the HPCI capacity is adequate to flood the pressurizer and sustain the solid system.

6) The stuck open relief valve was isolated much later after the reactor circulating pumps were tripped causing corresponding core damage. It is not known whether an isolation valve is provided ahead of the relief valve in this design. Also it appears that if this action of isolating the stuck relief valve had been taken before the primary pumps were tripped it is likely that the rest of the accident would not have taken the course as it had.

7) After the restoration of the emergency feedwater, recovery of secondary pressure in one of the steam generators is positive proof that before tripping of the primary pumps circulation was effective and the core was being cooled and the reduction of pressure from 1385 psi to 1050 psi was happening along the saturated line. This would have caused boiling in the core and circulation of voids in the primary lines ahead of the steam generator probably resulted in vibration in the pipe lines. The secondary pressure in the steam generator indicates effective heat transfer in the steam generator and hence it is unlikely that primary pumps would have cavitated. However, vibration in the lines seems to have disturbed the operator and probably as per instructions he had tripped the recirculating pumps.

8) Boiling in the core for a PWR under depressurisation and at low quality is advantageous to a certain extent because of increased heat extraction in boiling mode. A decrease in heat transfer occurs only when the flow regime changes to partial or stable film boiling. It is surprised that no core damage occured before the primary pumps were tripped. This surmise is confirmed by the fact that no activity was released through the containment before the tripping of the RCS pumps. The exact reason why the RCS pumps in both the loops were switched off needs investigation.

9) NRC Directives after the accident to PWR operators (reference cited earlier) was to include reactor scram on turbine trip, reactor scram coincidentally with pressuriser pressure and als o manual attention to feedwater emergency supply. The positioning of an operator without concurrent duties to watch emergency feedwater supply may not be adequate protection because of the likelihood of the operator sleeping off. More positive interlocking arrangements should be installed with indications available to the operator at the control panel.

1 0) Considerable discussion has arisen regarding thermosiphon cooling of the core. It is known that thermosiphon cannot perform without adequate heatsink. Also, it is essential to investigate the maximum power that can be rejected by the thermosiphon for a given plant and till such power level stabilise, additional cooling has to be ensured by other measures such as HPCI, LPCI etc.

11) Containment isolation philosophy has also been revised by NRC, i.e. not to wait for initial rise of pressure in the containment upto 4 or 5 psi before isolation. It would be interesting to know whether there are any instrumentation to detect the water activity in the containment sump which would isolate the containment on high activity.

12) Normally there should be a vent available on top of the vessel without which it should not have been possible to fill the system. It is necessary to understand why the vent was not ~~gxx~~ open to release the hydrogen bubble in the vessel which was causing considerable apprehension regarding flow blockage.

13) One hard decision was certainly open to the operating staff before stopping the primary coolant pumps to mitigate the effects of stuck release valve. This was to open all relief valves and let the system proceed along the full design LOCA path. Whether it was due to lack of authority or lack of perspective that prevented anybody to adopt this course of saving the reactor core, may be established.

14) The relief valve opening before the occurance of the pressure scram and remaining stuck flattens the slope of the pressure surge in the system thereby delaying the occurance of the scram. This evidently has been responsible for pumping in 9 to 12 fuel power seconds of energy in this case.

EVENT No.	Time	DESCRIPTION (N.R.C.)	REMARKS
1.	t = 0	LOSS OF CONDENSATE PUMP LEADING TO LOSS OF FEED WATER AND TURBINE TRIP	THE FEEDWATER FLOWS THROUGH COMMON HEADERS BEFORE AND AFTER THE POLISHING UNIT. WHICH IS THE VALVE WHICH CAUSED LOSS OF CONDENSATE PUMP IS NOT CLEAR. SIGNAL CAUSING TURBINE TRIP IS NOT ALSO CLEAR WHETHER IT IS LOW BOILER LEVEL, LOW BOILER PRESSURE OR LOW FEEDWATER FLOW.
2.	t = 3 to 6 SECS	ELECTROMATIC RELIEF VALVE OPENS (2255 PSI) TO RELIEVE PRESSURE IN RCS	FROM THE DATA GIVEN IT LOOKS AS IF THE LOGGING SEQUENCE HAS A 3 SECONDS CYCLE AND THE DATA SHOWS AN ANAMOLY AS IF THE RELIEF VALVES OPENED BEFORE HIGH PRESSURE SCRAM CAME IN. IT IS QUITE LIKELY THAT THE REACTOR DID SCRAM AT 2250 PSI BUT THE PRESSURE SURGED TO 2355 PSI WHEN RECORDED.
3.	t = 9 to 12 SECS	REACTOR TRIP ON HIGH RCS PRESSURE (2355 PSI)	
4.	t = 12 to 15 SECS	RCS PRESSURE DECAYS TO 2205 PSI (RELIEF VALVE SHOULD HAVE CLOSED)	THIS OCCURANCE IS NOT UNCOMMON. DESIGN FEATURES SHOULD CATER TO SUCH FAILURE.
5.	t = 15 SECS	RCS HOT LEG TEMPERATURE PEAKS AT 611°F, 2147 PSI (450 PSI OVER SATURATION)	- - - -

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EVENT NO.	TIME	DESCRIPTION (N.R.C.)	REMARKS.
6.	t = 30 SECS	ALL THREE AUXILIARY FEED WATER PUMPS RUNNING AT PRESSURE (PUMPS 2A AND 2B STARTED AT TURBINE TRIP). NO FLOW WAS INJECTED SINCE DISCHARGE VALVES WERE CLOSED.	CAUSE OF STEAM GENERATOR DRYOUT. HOLDUP OF ONCE THROUGH STEAM GENERATORS IS RATHER LOW BUT DETAILS NEED INVESTIGATION.
7.	t = 1 MIN	PRESSURIZER LEVEL INDICATION BEGINS TO RISE RAPIDLY	THIS CAN HAPPEN BECAUSE OF STUCK RELIEF VALVE RESULTING IN DEPRESSURISATION AND LOCAL VOIDS IN BULK FLUID. INDICATION MAY ALSO BE FAULTY.
8.	t = 1MIN	STEAM GENERATORS A AND B SECONDARY LEVEL VERY LOW DRYING OUT OVER NEXT COUPLE OF MINUTES.	- - - -
9.	t = 2 MIN	ECCS INITIATION (HPI) AT 1600 PSI	AS DESIRED FOR SMALL BREAKS.
10.	t = 4 to 11 MIN	PRESSURISER LEVEL OFF SCALE. HIGH ONE HPI PUMP MANUALLY TRIPPED AT ABOUT 4 MIN. 30 SEC SECOND PUMP TRIPPED AT ABOUT 10 MIN. 30 SEC.	OPERATOR ACTION ACCORDING TO NORMAL INSTRUCTIONS PROBABLY.

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EVENT NO.	TIME	DESCRIPTION (N.R.C.)	REMARKS.
11.	t = 6 MIN.	RCS FLASHES AS PRESSURE BOTTOMS OUT AT 1350 PSI (HOT LEG TEMPERATURE OF 584°F)	THIS HAS AN ADVANTAGEOUS QUENCHING EFFECT ON FUEL DUE TO INCREASED HEAT TRANSFER.
12.	t — 7 MIN 30 SEC	REACTOR BUILDING SUMP PUMP CAME ON	THE REASON IS NOT CLEAR SINCE THE QUENCH TANK IS REPORTED TO HAVE BLOWN AT 190 PSI AT 15 MINUTES.
13.	t = 8 MIN	AUXILIARY FEEDWATER FLOW IS INITIATED BY OPENING CLOSED VALVES.	----
14.	t = 8 MIN 18 SEC	STEAM GENERATOR B PRESSURE REACHED MINIMUM	----
15.	t = 8 MIN 21 SEC	STEAM GENERATOR A PRESSURE STARTS TO RECOVER	THIS IS A POSITIVE INDICATION THAT RCS FLOW IS AVAILABLE AND NO CORE OR NOZZLE WAS EXPOSED IN THE REACTOR VESSEL.
16.	t = 11 MIN	PRESSURIZER LEVEL INDICATION COMES BACK ON SCALE AND DECREASES	AT LEAST THE ENERGY SURGE INTO RCS IS BEING ACCOUNTED FOR IN THE REESTABLISHED HEAT SINK AND SUBCOOLED FLUID ENTERS THE REACTOR.

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EVENT NO.	TIME	DESCRIPTION (N.R.C.)	REMARKS.
17.	t = 11 to 12 MIN	MAKEUP PUMP (RCCS HPI FLOW) RESTARTED BY OPERATORS.	- - - -
18.	t = 15 MIN	RC DRAIN/QUENCH TANK RUPTURE DISK BLOWS AT 190 PSI (SETPOINT 200 PSI) DUE TO CONTINUED DISCHARGE OF ELECTRO-MATIC RELIEF VALVE.	REFER TO EVENT No.12
19.	t = 20 to 60 MIN	SYSTEM PARAMETERS STABI-LISED IN SATURATED CONDITION AT ABOUT 1015 PSI AND ABOUT 500°F.	DECAY POWER IS ABOUT 3 PER CENT OF FULL POWER AT THIS PERIOD. AN EQUILIBRIUM SEEMS TO HAVE BEEN ESTABLISHED FROM EVENT NO.11 AND EVENT NO.13. HOWEVER SINCE PRESSURE IS DECRADING ON THE SATURATED LINE AND HOT LEG TEMPERATURE PEAKING IS NOT EVIDENT, NO CORE FLOW STAGNA-TION OR EXPOSURE IS EVIDENT. THE VOIDS IN THE BULK CIRCULATING FLUID CAN CAUSE NOISE AND VIBRATION IN RCS, AS HAS BEEN REPORTED.
20.	t = 1 hr. 15 mt.	OPERATOR TRIPS RC PUMPS IN LOOP B	THE RESULT OF DEGRADATION OF FLOW ON THE EQUILIBRIUM ESTABLISHED EARLIER AND BETWEEN NOW AND EVENT 21 IS NOT INDICATED. IT IS LIKELY THAT THE NOISE AND VIBRATION IN RCS COULD HAVE INCREASED DUE TO INCREASE IN VOIDS. THE TRIPPING OF THE RCS PUMPS IS JUSTIFIED AS A CONSEQUENCE. BUT AS CAN BE SEEN THERE-AFTER, THE EFFECT HAS BEEN DISASTROUS TO THE CORE.
21.	t = 1 hr 40 mt.	OPERATOR TRIPS RC PUMPS IN LOOP A	

EVENT NO.	TIME	DESCRIPTION (N.R.C.)	REMARKS
22.	t = 1¾ hrs to 2 hrs.	CORE BEGINS HEAT UP TRANSIENT - HOT LEG TEMPERATURE BEGINS TO RISE TO 620°F (OFF SCALE WITHIN 14 MINUTES) AND COLD LEG TEMPERATURE DROPS TO 150°F. (HPI WATER)	THIS INDICATES THAT THERE IS COMPLETE FLOW STARVATION IN CORE, AND THAT THE TOP OF THE CORE IS SUPERHEATING THE STEAM AND LIKELY CORE EXPOSURE HAPPENED WITH DAMAGE TO FUEL. THE RATE OF POWER GENERATION IS ABOUT 1.5 PER CENT OF FULL POWER AND IF COMPLETE LOSS OF COOLING IS ASSUMED THE FUEL TEMPERATURE STARTS RISING AT ABOUT 3°F/Sec. TO REACH CLAD RUPTURE AND INITIATION OF METAL WATER REACTION IT WOULD TAKE APPROXIMATELY 300 SECONDS. THE THERMAL CAPACITY OF FUEL ALONE IS ABOUT 11280 Bhu/°F AND HEAT GENERATION IS 39400 Bhu/Sec. THE RCS FLOW WAS NOT RESTARTED TILL MUCH LATER. (EVENT 34 AT 16 HOURS)
23.	t = 2.3 hrs.	ELECTROMATIC RELIEF VALVE ISOLATED BY OPERATOR AFTER S.G.-B ISOLATED TO PREVENT LEAKAGE	IT IS INEXPLICABLE WHY THIS WAS NOT DONE BEFORE RCS PUMPS WERE TRIPPED.
24.	t = 3 hrs.	RCS PRESSURE INCREASED TO 2150 PSI AND ELECTROMATIC RELIEF VALVE OPENED	THE REST OF THE TRANSIENT ONLY INDICATES THAT THE REACTOR WAS BLOWING CONTINUOUSLY INTO RCS DRAIN TANK AND THEN TO CONTAINMENT. THE RELIEF VALVES WERE ULTIMATELY OPENED TO REDUCE THE PRESSURE TO 500 PSI TO INITIATE CORE FLOODING.

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EVENT NO.	TIME	DESCRIPTION (N.R.C.)	REMARKS
24. contd.			THIS COULD HAVE BEEN DONE MUCH EARLIER AT EVENT 20 IMMEDIATELY AFTER TRIPPING RCS PUMPS TO GO THROUGH THE DESIGNED PATH OF A FULL 'LOCA' AND SAVE THE CORE AND ACTIVITY RELEASE. THIS IS HOWEVER A DIFFICULT DECISION TO TAKE.