UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 8 1981

AEOD/E115

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MEMORANDUM FOR: Carlyle Michelson, Director
Office for Analysis and Evaluation of Operational Data

FROM: Harold L. Ornstein
Office for Analysis and Evaluation of Operational Data

SUBJECT: ADDITIONAL INFORMATION ON EVENTS AT TMI-2 DURING PREOPERATIONAL TESTING (SEPTEMBER 5-12, 1977)

Subsequent to our discussion with Bill Dircks on this subject, we received additional logbook sheets. The additional logbook sheets reveal how the steam bubble formed (shutdown of all four RCPs). In response to my request, John Pellet has put together the highlights of the event. A copy of the highlights is enclosed.

The logbook reveals the occurrence of a significant precursor event which appears to be "more reportable" than the discovery of the presence of a steam bubble in the primary system. There was an interaction between a secondary side component and primary side safety-related components, i.e., failure of the condensate polishers caused resin to clog the strainers from the nuclear service closed cooling water pumps, thereby causing a loss of RCP seal cooling, RCP motor cooling, and makeup water pump motor cooling (shift #1 on 9/7/77).

H L Ornstein

Harold L. Ornstein
Office for Analysis and Evaluation of Operational Data

Enclosure:
As Stated

cc w/enclosure:
J. Heltemes
J. Pellet

J. Pellet

Overview of TMI-2 Log, September 5 - 12, 1977

1. 9-5-77 shift 1 — High differential pressure across condensate polishers causes condensate booster pump trip on low suction pressure so polisher bypass opened and polisher repair started.

2. 9-5-77 shift 3 — Polisher back on line, filling demineralized water tank (from condensate system).

3. 9-7-77 shift 1 — All Nuclear Service Closed Cooling Water (NSCCW) pumps tripped due to clogged pump suction strainers. Clogging resin came from condensate polishers thru demin. water system. Resin found in demin. water system in turbine and auxiliary buildings. Loss of NSCCW produced a loss of RCP seal and motor cooling. Also loss of cooling to makeup (MU) pump motors occurred and makeup was secured. All RCP's were immediately secured on loss of both seal injection and seal cooling. Nuclear Service River Water cooling to MU pumps was established and MU restored.

4. 9-7-77 shift 2 — Isolated normal pressurizer spray line to work on liquid waste system valving (note: spray lost when RCP's secured).

5. 9-7-77 shift 3 — RCS on natural circulation with makeup via MU-P-1B. RCS temperature and pressure decreasing. Continuously filling and draining pressurizer to reduce RCS pressure. Secured condensate booster and main feedwater pumps due to plant cooldown (1 CBP on).

6. 9-8-77 shift 1 — Pressurizer level constantly surging. Started venting pressurizer to reactor drain tank.

7. 9-8-77 shift 2 — RCS at 200°F and 400 psig. Pressurizer level being restor[ed] to normal level. Pressurizer level increased 150" while

venting RCS from 500 psig to 460 psig. Pressurizer temperature about 340° F. From the log, "Apparently the reference legs have flashed and there was no steam in the pressurizer to fill the reference legs." (note that at 460 psig Tsat > 450°F).

8. 9-8-77 shift 3 — RCS at 250°F and 340 psig. Pressurizer level transmitter LT-3 backfilled. LT-1, LT-2, and LT-3 all read the same in the control room and are believed to be correct. DHR started with B pump with RCS at 160°F and cooled to 100°F. Whenever the pressurizer was vented pressurizer level increased. DHR to vessel isolated and maximum auxiliary pressurizer spray established to try to cool the pressurize Secured seal injection and return on all RCP's.

9. 9-9-77 shift 1 — RCS at 150 psig (Tsat ≅ 360°F). Closed pressurizer vent and, "...applied nitrogen to the pressurizer. The pressurizer level came down proving that there was a steam bubble in each of the hot legs." Left nitrogen on until pressure started to increase then secured. Opened cold leg drains to Reactor Coolant Bleed Tank and drained RCS until pressurizer level recorder was not decreasing. The hot legs and pressurizer were equalized in pressure thru the nitrogen piping. RCS pressure decrease continued by venting hot legs to atmosphere and pressurizer to Reactor Drain Tank. Secured cooldown with RCS at about 100°F. Connected Tygon tubing to RCS to measure RCS level when pressure decreased sufficiently.

10.	9-9-77 shift 2	RCS vented and drained such that level in tubing decreased from 92" to 22".
11.	9-9-77 shift 3	Secured makeup when tubing indicated a level of 139". Opened hot leg and pressurizer vents to atmosphere and drained RCS to an indicated 16".
12.	9-10-77 shift 2	Inside containment mitrogen regulator (5#) found "blown apart" (NM-V142).
13.	9-12-77 Review	"There is no reason given for how we got into a problem on pressurizer level. A change to cooldown procedure could be made if we knew what to do."

DISCUSSION OF TECHNICAL INFORMATION AVAILABLE ON THE SEPTEMBER 1977 EVENT

The event during the September 1977 hot functional test at TMI-2 is potentially significant because there were indications of primary system voiding. The presence of such voids could challenge, and possibly prevent, decay heat removal via natural circulation. The assurance of natural circulation is a design requirement since it may be necessary as a result of loss of power conditions. The evidence presented below was compiled from the pertinent sections of the TMI-2 operations log.

The presence of primary system voiding is indicated by several entries in the log. For example, the pressurizer level changed dramatically while venting the pressurizer to reduce Reactor Coolant System (RCS) pressure. This would indicate an expanding void in the RCS outside the pressurizer. Additionally, when nitrogen was injected into the pressurizer steam space, the pressurizer level went down although pressure did not increase as expected. This would occur if the RCS were not solid and the void was being compressed. Further, there were indications that the pressurizer was not controlling RCS pressure which again suggests voiding. On the other hand, the pressures and temperatures given in the log indicate that a high degree of subcooling in the RCS was maintained which normally would indicate an absence of steam bubbles or steam voiding.

If voiding was present in the RCS, the voiding could have been caused either by the presence of steam, i.e., a steam bubble or by a gas such as nitrogen. As noted earlier, a large subcooling margin (in excess of 100°F) appears to have been maintained. Thus, even though the operation log specifically states that a steam bubble had formed, the evidence of a steam bubble does not appear to be conclusive. Further, there is no direct evidence that any gas voids were present but nitrogen is involved with the RCS equipment and its presence would appear to be the next most likely cause for voiding.

The nature, size, and location of the voids, if present, cannot be conclusively determined from the evidence examined. Furthermore, the critical question concerning whether this event could have occurred after fuel loading could not be determined from the operating log information. In fact, it could not be verified that natural circulation was established after loss of forced flow or whether heat was being removed from the steam generators by the secondary system. The system configuration was not described in sufficient detail to permit identification of unique startup or test equipment and operating procedures. Consequently, it could not be determined whether this event could have occurred after fuel load.

An additional factor noted in the operating log was the apparent interaction of the condensate polishers with the Nuclear Service Closed Cooling Water System. The latter system performs safety-related functions by removing heat from the components in several safety systems, and its operation is necessary for reliable, long-term plant operation. The interaction between safety and nonsafety systems could also be potentially significant in terms of the safety assurance for the plant. Accordingly, this problem could also be reportable under the same regulations as discussed in terms of the voiding condition. Again, however, the evidence in the operations log is not conclusive. It could not be determined that this interaction could occur during power operation rather than it being exclusively a startup or preoperational problem.

Significant deficiencies identified during the design and construction phase are required by NRC regulations to be reported (10 CFR 50.55(e)). At the time of the September 1977 event, the plant was in the construction phase, i.e., in hot functional tests with no fuel present. As discussed in Enclosure X, we have reviewed the available information concerning this event, but we find the information and data to be quite limited and in some cases inconsistent.

It is clear that this event would be reportable if ~~this event~~ it could have occurred after fuel load and if the conditions would have adversely affected the safety of operations; for example, the ability to cool the core by natural circulation. However, in view of the conflicting and limited information on this event, it has not been possible for us to define with confidence exactly what did occur and whether this event could have occurred after fuel load. Consequently, it is also not possible to state unequivocably that this event was or was not reportable. We would add, however, that it is clearly in the best interests of the licensee, the industry, and the NRC to have all events reported which could potentially affect the safety assurance of nuclear plants, regardless of the strict legal requirements for reporting.

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10 CFR 50.55(e) requires reporting of significant deficiencies found in construction and design during the construction and preoperational phase. At the time of the September 1977 event, the plant was in hot functional testing with no fuel present. We have reviewed the available logs, strip charts, and other information and found that, while the instrumentation satisfied the requirements for that testing, we are unable to reconstruct the details of the event. Thus, a firm conclusion regarding reportability is prevented. Nonetheless, the event may have had ramifications with respect to the plant's ability to function in the natural circulation mode once a core had been loaded and decay heat was available. Had this been recognized at the time, we would probably regard the event as reportable, depending on the details. Notwithstanding the strict legal requirements for reporting, the Commission's policy is to be informed of such events to improve its overall understanding of the operational characteristics of nuclear reactor plants.