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October 11, 1989
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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Loss of Decay Heat Removal Event of June 27, 1988

The purpose of this letter is to fulfill a commitment made to the NRC during Inspection 88-13 to provide a letter documenting the details of an event that occurred on June 27, 1988 which resulted in a loss of Decay Heat Removal (DHR) cooling for approximately 13 minutes.

This event was not reportable in accordance with 10CFR 50.72 or 10CFR 50.73. This voluntary report is being submitted at the request of the NRC. However, GPUN does not believe that it would be appropriate to increase the scope of its reporting requirements beyond the regulations as a formal commitment.

A copy of the Plant Incident Report No. 1-88-03 was provided to the NRC on-site resident inspector's office along with a GPUN Human Performance Evaluation System (HPES) Report, entitled "Inadvertent Isolation of Decay Heat Removal Pathway During Instrument Calibration", dated August, 1988. The event was discussed in the TMI-1 Resident Office monthly status report for the period June 4, - July 9, 1988 and NRC Inspection Report 88-13, dated August 26, 1988.

This event was a result of personnel error by an I&C technician while performing a calibration of the High Pressure Injection (HPI) and Low Pressure Injection (LPI) channels of the Engineered Safeguards (ES) System as required by TMI-1 Technical Specifications. There were no adverse consequences from this incident. Public health and safety were unaffected. The event did not go beyond the design basis of the DHR System. Operator response to the loss of DHR was timely and appropriate.

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Plant Conditions:

At the time of the incident the plant was in cold shutdown and the Reactor Coolant System (RCS) had been drained to approximately 15" above the cold leg center line to support the 7R Refueling Outage work activities. Decay Heat was being removed using the "A" DHR loop and the "B" DHR loop was operable. Surveillance Procedure 1302-5.8 "HPI/LPI Analog Channel Calibration" was being performed. There were no other outage activities that had any direct impact on this incident.

Sequence of Events:

At approximately 10:40 AM on June 27, 1988 two licensee I&C Technicians had just completed a calibration of the RCS pressure transmitter (RC3A-PT4) instrument string per procedure (1302-5.8). While restoring the string to normal per Section 8.2.3 of that procedure, the technician at the ESAS Cabinets was performing the cabinet related operations, utilizing the sign-off copy of the procedure. The other technician was in the Reactor Building applying pressure to the transmitter and performing activities in the Reactor Building associated with the evolution as directed by the technician at the ESAS cabinets. The event occurred when steps 8.2.3.1, 8.2.3.2, and 8.2.3.3 of the procedure were performed out of sequence. Step 8.2.3.3 was performed before 8.2.3.2. This caused DH-V2 (Decay Heat Drop Line Isolation Valve) to shut, isolating the suction of the Decay Heat Removal Pump (DH-P1A). Control Room personnel recognized the alarms for DHR system low flow and DH-P1A high vibration alarms and secured the pump.

Control Room personnel were immediately dispatched to determine the condition of the pump (DH-P1A) and also to investigate the potential for performance of the ESAS test procedure as the source of the problem. It was quickly established by local observation of the ESAS cabinets that DH-V2 had stroked closed due to the 400 psig interlock which was being tested by the procedure (1302-5.8). The interlock condition (a tripped 400 psig bistable) was reset and DH-V2 was opened. DH-P1A was determined to be undamaged by local observation, was vented and restarted at approximately 10:53 AM. During the period of time in which there was no DHR flow, (approximately 13 minutes) the reactor vessel water temperature increased from approximately 123°F to 155°F as read on the Backup Incore Read out (BIRO).

Discussion:

The incident investigation process revealed that both technicians on this job had a good understanding of the procedure. They were aware of the steps involved, the proper step sequence, and the ramifications of performing these steps improperly. The technician at the ESAS cabinets had performed a meter-channel comparison as directed by step 8.2.3 and had become involved in a discussion about a difference in the readings among the three channels. This discussion involved the two individuals on the job and a third individual, also an I&C technician, in the vicinity of the cabinets. After the issue was resolved in this discussion, the individual in the Reactor Building went off the headset to move the equipment to the remaining channel.

The technician at the ESAS cabinets then performed step 8.2.3.3 which removed the bypass from DH-V2, before performing step 8.2.3.2 which resets the 400 psig bistable. The technician took this action despite the existing note after step 8.2.3.2 which describes the adverse condition which occurred.

The cause of this event was personnel error. Even though the I&C technician was attempting to follow each step in the procedure in its proper order, steps were performed out of sequence. It was determined that the technician had familiarized himself with the actions that were necessary and the ramifications of improper performance. However, the individual became preoccupied with a previous portion of the test. Consequently, when he returned to the steps in question, he lost track of the prime objective of this section of the procedure, to restore normal operation without closing DH-V2. He did not follow the proper sequence and he failed to take note of the existing procedure precautionary information.

Safety Assessment:

This event did not pose a nuclear safety concern. The core remained covered with a minimum subcooling margin of 60°F without the need to employ additional means of cooling or water addition. The Borated Water Storage Tank (BWST) provides a large reservoir of water as a heat sink. The BWST was available for core cooling in accordance with Emergency Procedure 1202-35, "Loss of Decay Heat Removal System" if needed.

At the time of the event the plant had been shutdown for about ten days. Based on generic decay heat generation curves for TMI-1, the core was producing approximately 4.8 Mwt of decay heat (0.18% full power). Similar curves for plant heatup with the RCS drained down shows that saturation is reached after 30 minutes with no means of DHR cooling in use. From testing during the 6R refueling outage, actual plant data showed a heatup rate less than that predicted by the curves. At the decay heat level present during this event it would take approximately 16,900 lbm/hr of water boil off at atmospheric pressure to remove the decay heat. With a level of 16 inches in the cold leg piping, it would take approximately 2½ hours to boil down to within 12 inches of the top of the core without any makeup. These times are given for reference only assuming that DHR flow were lost for an extended time and no other measures were taken to cool the core.

The need for the 400 psig interlock on DH-V1 and DH-V2 has been addressed previously by GPUN. The interlock is needed to protect the low pressure DHR System piping from exposure to high pressure. The interlock is designed to prevent inadvertent valve opening above 400 psig, which is the major concern. It is also designed to shut DH-V1 and DH-V2 should re-pressurization occur while operating in the DHR cooling mode. Adequate procedural controls are in place to minimize inadvertent valve cycling of DH-V1 and DH-V2 and adequate guidance is provided to recover DHR forced cooling should the isolation valves go closed inadvertently for any reason.

Corrective Actions Taken as a Result of This Event:

Instrument technicians have been reminded that this event emphasizes the necessity for proceeding through each step of procedures like this one, methodically reading a step, performing the step and then signing-off on that step before going on to the next one.

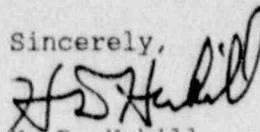
Surveillance Procedure 1302-5.8 was revised to ensure that the DH-V1 and DH-V2 breakers are open while testing their respective bistable interlocks (Reference: Rev 13, effective 10/21/88).

Conclusion:

There were no adverse consequences from this event. This event did not go beyond the design basis of the DHR System and did not result in a total loss of cooling capability because the BWST was available to provide a large reservoir of water for core cooling, in accordance with Emergency Procedure 1202-35, if needed. The operator response to this event was timely and appropriate. Forced flow DHR cooling was restored and RCS temperature increased by about 32 F° to approximately 155°F, well below the boiling point of 212°F. The changes to Surveillance Procedure 1302-5.8 described above will prevent this particular event from recurring. Adequate procedural controls are in place to minimize inadvertent valve cycling of DH-V1 and DH-V2 and adequate guidance is provided to recover DHR forced cooling should the isolation valves go closed inadvertently for any other reason.

The procedural guidance that has been incorporated, the knowledge gained from this event, and the training of operators in this area will further decrease the likelihood of any loss of DHR event and serve to mitigate the consequences such a should loss occur for any reason.

Sincerely,



H. D. Hukill

Vice President and Director, TMI-1

HDH/MRK

cc: J. Stolz
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