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Office of Nuclear Reactor Regulation
Attn: J. F. Stolz, Director
PWR Projects Directorate No. 6
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Emergency Feedwater Block Valves
(NUREG 0737 Item II. E.1.2)

As part of the ongoing review of the long term Emergency Feedwater (EFW) System, GPUN has performed several analyses and evaluations of the EFW block valves (EF-V52, 53, 54 and 55). Also, GPUN has reviewed events which have recently occurred at other plants involving motor operated valves. Based on these analyses/evaluations, GPUN has determined that the motor operators for the EFW block valves are not required for design basis events.

The design purpose of those valves in the fully safety grade EFW system was to provide isolation of EFW flow to the affected steam generator under a Main Steam Line Break (MSLB) or Feedwater Line Break (FWLB) coincident with a failure in the open position of the EFW control valve. Also considered were Loss of Feedwater (LOFW) and Loss of Offsite Power (LOOP) events. A summary of the analyses of these accidents under worst case single failures demonstrates the capability of the EFW System to provide EFW to both Once Through Steam Generators (OTSGs) or to provide EFW to only the unaffected OTSG (see the attachment to this letter for the summary of the analyses).

In all cases, the worst case single failure was an open failure of one EFW flow control valve (EF-V30). The control valves for the fully safety grade EFW system will be modified to fail closed on loss of air or control power. There will be no single failure that will cause more than one flow control valve to fail open. In each case, the break location is either inside the Reactor Building or inside the Intermediate Building. Breaks inside the Reactor Building cause a more severe plant response, while breaks inside the Intermediate Building make entry to the building for local operation of the EFW block valves, in the time required to mitigate the accident, impossible. In this case, other actions can be taken in the time required to mitigate the accident.

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For breaks inside the Reactor Building, isolation of EFW flow to the affected OTSG is accomplished by manual operation of the EFW block valves. For breaks inside the Intermediate Building, isolation is accomplished by tripping the motor driven EFW pump and closing the appropriate discharge header valve (EF-V-2A or 2B). Flow control can then be accomplished by reopening/closing the discharge header valve.

It has always been the design philosophy of GPUN to simplify both the design and operation of systems wherever possible, provided that the licensing basis is maintained and the inherent safety function of the system is not degraded. The change from remote manual valve operation to local manual valve operation accomplishes this design philosophy.

Sincerely,

H. D. Hukill
Director, TMI-1

HDH/JA/jh:3133f

cc: R. Conte
J. Thoma

Attachment

DESIGN BASIS ACCIDENT SUMMARY FOR MANUAL EFW BLOCK VALVES

1. STEAM LINE BREAK ACCIDENT

Steam line break accidents inside the Reactor Building must be controlled so that the fuel is not damaged and so that the containment design pressure is not exceeded. This accident was extensively analyzed in the TMI-2 FSAR (Appendix 15B) where acceptable results were demonstrated with EFW flowrates of 1250 gpm, which is greater than the TMI-1 system capacity. If the control valve for the affected OTSG failed to open, the event would be mitigated even without operator action. Therefore, closure of the motor operated block valve is not required for event mitigation. It should be noted that use of the TMI-2 FSAR analysis for this event is acceptable because the TMI-1 system response is bounded by the TMI-2 system response.

If the failure of a control valve occurred on the unaffected OTSG, plant cooldown could still be accomplished after the affected OTSG was isolated. With all three EFW pumps operating, the flow to the intact OTSG is approximately 600 gpm. This will result in an acceptable cooldown rate (see the discussion below on the LOFW transient for substantiation of this statement). If the break is inside the Reactor Building, the operator has access to the block valve, and it can be closed thereby returning automatic level control via the redundant control valve. In the interim, if a slower cooldown rate is desired, one EFW pump can be tripped by the control room operator. This will result in a flowrate of approximately 440-600 gpm to the intact OTSG (depending on whether the turbine or motor-driven pump is tripped).

For the break inside the Intermediate Building, containment pressure is not a concern, but isolation of the break must occur within one hour in order to satisfy the analytical assumptions used in developing the environmental qualification temperature and pressure profiles for the Intermediate Building. Since the block valves are inaccessible due to the building environment, the control room operator must isolate EFW flow to the affected OTSG by closing the appropriate EFW sectionalizing valve (EF-V2A or 2B) and tripping the motor driven EFW pump aligned to the affected OTSG. Both of these actions are initiated from the control room so that one hour is ample time to isolate the break.

2. FEED LINE BREAK ACCIDENT

Feed line break accidents are events which result in inadequate primary to secondary heat removal. Acceptance criteria for this event are that thermal power is limited to less than 112% (resulting in DNBR greater

than 1.3) and RCS pressure is less than 2750 psig. Open failure of the EFW control valve increases the EFW flowrate and minimizes the overheating transient. In the long term, plant cooldown would be accomplished in the same manner in which the plant would be cooled down after a steam line break.

If one motor driven pump has to be tripped to isolate EFW flow to the affected OTSG, then plant cooldown could be accomplished with the remaining motor driven EFW pump (there is no identified single failure that will cause the loss of a motor-driven EFW pump and the open failure of a control valve). Cooldown at low OTSG pressure/temperature will be slowed with only one EFW pump because of the reduced steam generator steaming capacity, however, core cooling is unaffected. Eventually, access to the Intermediate Building would be regained and the block valve would be closed. This would allow for restoration of the second EFW pump and for a more rapid cooldown rate.

3. LOSS OF OFFSITE POWER

Except at very low decay heat levels, the cavitating venturies sufficiently limit flow so as to prevent overcooling events from occurring. A failure of the level controller during a loss of offsite power was analyzed using the computer program RETRAN and presented in the TMI-1 Restart Report (Appendix 8A, Case 5). The results of the analysis are more severe than the failure of one EFW control valve. For the single failure of one EFW control valve, the operator can throttle flow to one OTSG and limit the EFW flow to nearly half the flow indicated in the TMI-1 Restart Report, Case 5. In Case 5, flow to both OTSGs is uncontrolled, however, there is still sufficient time to allow for operator action. For Case 5, the RCS cooldown rate is between 5°F and 6°F per minute. At this rate, the RCS would cool down below 500°F at approximately 700 seconds following the reactor trip. The analysis performed in the Restart Report used an ANS decay heat multiplier of 0.01. Therefore, if this type of event were to occur any time after the plant had been in full power operation for several days, the operator would have significantly more time in which to take action.

4. LOSS OF FEEDWATER

The response to a loss of feedwater simultaneous with the failure of one control valve, assuming no other equipment failures, was evaluated. One "A" EFW control valve is assumed to go open and fail in that position while level control on the OTSG-B works properly, throttling flow when the level setpoint of 30 inches is reached.

Because of the flow limitation accomplished with the cavitating venturies, the effect of a failed open control valve is minimal. Flow to the OTSG-B is expected to throttle back in less than 10 minutes as steam generator level reaches the control setpoint of 30 inches. The OTSG-A level continues to increase due to the failure of the control valve, however, the combined EFW flowrate is only 600 gpm. This flowrate exceeds the decay heat requirements, but is sufficiently low so as to prevent severe overcooling.

This event was analyzed using a decay heat multiplier of 0.9 to conservatively bound a trip within one day after achieving full power. The dominant source of heat for the first several hours after a reactor trip is from short-lived radioisotopes which reach equilibrium in the first several days of a fuel cycle. Consequently, the decay heat load is high for the first several hours of the overcooling transient.

For a nominal trip response from full power, there will be virtually no overcooling. For the bounding case presented in the TMI-1 Restart Report, the operator still has in excess of 10 minutes to respond.