UNITED STATES
NUCLEAR REGULATORY COMMISSION
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MEMORANDUM FOR: D. F. Ross, Jr., Director, Bulletins and Orders Task Force

FROM: B. A. Wilson, Systems Group, Bulletins and Orders Task Force

SUBJECT: REACTOR COOLANT PUMP TRIP REQUIREMENTS FOR B&W PLANTS (IE Bulletin 79-05C)

The purpose of this memo to document my concerns regarding the requirement to immediately trip the reactor coolant pumps (RCP's) in the event of a reactor trip and safety injection initiation caused by a low pressure condition. As was discussed during several meetings we have had on this subject, my concern is only wth B&W designed plants due to the close coupling between the primary and secondary systems. For purposes of documentation, the chronology of events that led to this requirement will be presented and then to my concerns for non-LOCA events.

Immediately following the TMI-2 accident, the staff recognized that had the RCP's been kept running, significant core damage may have been averted. Therefore, I&E Bulletins, 79-05 and 79-06 directed the licensees to revise their emergency procedures to require that in the event of HPI initiation, at least one RCP per loop (differs depending on vendor design) shall remain operating.

In subsequent analysis requested by the staff, B&W calculated that for a certain spectrum of primary system break sizes, the possiblity exists of exceeding 10 CFR 50.46(b)(1) criteria (2200°F peak clad temperature) if the RCP's are tripped at a particular time interval into the event. The range of break sizes examined for the analysis was from 0.025 ft^2 to 0.2 ft^2 and were located in the pump discharge piping. The effect of tripping the RCP's was analyzed when the primary system reached 90% void fraction since B&W expected this condition to produce the highest peak clad temperature (PCT). Further, Appendix K assumptions were used (as required by 10 CFR 50.46) which included 1.2 times the ANS Standard for decay heat rate. In a subsequent analysis, B&W included a curve (Figure 11) which indicated a critical region in which tripping the RCP's would result in the unacceptable PCT. This region varied from approximately 200 seconds to 4000 seconds depending on break size.

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In summary, the B&W analysis showed that if a small break occurred and was between 0.025 ft^2 and 0.2 ft^2 and was on the pump discharge piping and if the reactor coolant pumps were tripped during a certain time interval, then using the conservative Appendix K assumptions, the PCT of the fuel may exceed 2200°F.

As a result of this analysis, B&W notified the utilities with B&W plants that they recommended immediately tripping all operating RCP's upon reactor trip and high pressure injection caused by low reactor coolant system pressure. A meeting was held in Mr. Edson Case's (Deputy Director, NRR) office on July 20, 1979 to discuss how the NRC should respond to the B&W findings. Although several of us recommended further analysis to determine the effects of the pump trip requirements, the decision was made to issue I&E Bulletins 79-05C and 79-06C. The latter bulletins, in effect, imposed the RCP Trip requirements, as analyzed by B&W, on Westinghouse and Combustion Engineering designed plants. In addition to requiring the RCP trip, the bulletin requested further analyses and guidelines concerning the impact of the RCP trip requirements for both LOCA and non-LOCA transients.

This brings us to the present time in which all utilities with operating B&W plants have committed to comply with Item 1.A of Bulletin 79-05C. We must assume that any event which causes high pressure injection due to low reactor coolant system pressure will result in the operator tripping the RCP's. Now let me present the probable consequences of this RCP trip requirement for non-LOCA events.

In response to IEB 79-05C, B&W analyzed the impact of a RCP trip on an overcooling event, namely a 12.2 ft^2 steam line break. The results of the analysis indicate a primary system void volume of about 400 ft^3 with natural circulation "temporarily reduced." This is a result of the accident and tripping the reactor coolant pumps. Further into the event, it was calculated that pressurizer level would be restored and system pressure turned around from HPI flow after approximately 14 minutes. During the transient, however, the operators would be faced with an empty pressurizer, the hot leg at saturation temperature and a primary ΔT of almost 200°F for five minutes or more (see Figure 3.16 Supplemental Small Break Analysis).

At the request of the Analysis Branch, DSS, calculations of an overfeeding event were performed at Brookhaven National Laboratory. The transient assumed a turbine-trip-reactor-trip with full continuous main feedwater flow and subsequent RCP trip after HPI initiation. The results indicated 30 ft^3 of voiding in the hot leg without interruption of natural circulation. The calculation did not include introduction of cold auxiliary feedwater into the upper portion of the steam generators after tripping the fourth RCP. The analyses performed so far, supports the position that the RCP trip requirement will lead to primary system voiding with uncertain effects on natural circulation for non-LOCA events.

What, then are the probabilities of non-LOCA events which result in HPI initiation? The recent event at Davis-Besse (September 23, 1979) came very close to HPI initiation. Following a reactor trip, an extended safety valve blowdown caused a primary system depressurization and loss of pressurizer level indication of approximately one minute. Had the pressure dropped sufficiently to cause HPI initiation (one of the two SFAS channels did in fact trip) the following would probably have happened:

1. Operator follows procedures and trips the four RCP's
2. Auxiliary feedwater pumps automatically start on trip of fourth RCP and deliver cold feedwater into upper section of steam generators.
3. Operator allows auto-essential level control system to fill OTSG's to 96" (as required by procedures).
4. Additional overcooling of primary shrinks level out of pressurizer and causes void formation in hot legs (hot legs at saturation temperature).
5. Due to HPI and steam formation in primary system, pressure increases above shut off head of SI pumps ($\sim$ 1700 psig) resulting in no injection of water into RCS.
6. If there was no or inadequate natural circulation, system pressure would rise to lift setpoint of PORV and/or safety valves.
7. Operator could reset SI signal and start charging pumps. However, these are low capacity and could only deliver about 200 gpm at rated system pressure.
8. If an RCP could not be restarted, B&W Small Break guidelines direct the operator to open the PORV to decrease pressure to 100 psig above secondary pressure. Repeat as necessary, until natural circulation is established.

The primary system at this point has a remarkable similarity with the TMI-2 accident, i.e., hot legs voided at saturation temperature, little or no makeup flow to RCS, and PORV open.

The point of the preceding discussion is that this is not an unlikely transient. The September, 1977 event at Davis-Besse should be evidence enough. Actuation of HPI on low pressure is estimated to have an occurrence frequency of 0.5 to 1.0 per reactor year. An incident at Oconee Unit 2 is attached as another example. A loss of ICS auto power caused by a circuit breaker pulling loose, resulted in high pressure injection with the primary system reaching a minimum pressure of 1450 psig. It is also noted on page 2 that they believe the PORV opened due to an erroneous signal as primary system pressure was rapidly decreasing. If the RCP's had been tripped during this event I feel certain that this would have had primary system voiding.

Incidents of this type apparently place the B&W plants in the undesirable situation of being unsafe if the RCP's are tripped or if they are not tripped.

I believe there is a solution however, In the long term, an automatic pump trip on a coincidence signal of HPI and RCP power or current may be the solution. This should however, be fully explored before being implemented.

In the near term, I believe the emergency procedures should be modified to require the RCP trip only after pressure has decreased below 1200 psig. The B&W analysis indicates primary system pressure to be less than 1200 psig within one minute for the smallest of the breaks considered to be a problem (.025 ft^2 -Figure 2-3). Also, if Figure 2-5 of the original B&W analysis is compared with Figure 11 of the supplemental analysis it can be seen that the larger breaks require an earlier trip of the RCP's but primary system pressure will be lower. For example, a 0.2 ft^2 break requires the pumps to be tripped within 200 seconds but by that time the RCS pressure will be about 1000 psig leaving little doubt to the operator that there is indeed a break. The pressure for a smaller break will level off at secondary system pressure, about 1100 psig or higher which may cause some confusion as to whether or not the pumps should be tripped but the critical region doesn't begin until about 400 seconds into the event. For a 0.025 ft^2 break, the critical time period is in the order of 40 minutes.

To further support the 1200 psig pump trip requirement a review of all transients at B&W plants should be immediately performed to determine the frequency of occurrences in which the RC system depressurized to less than 1200 psig. Based on conversations with several utility operations personnel, such incidents have occurred but are relatively infrequent compared with safety injections due to low pressure.

In conclusion, I recognize that the RCP trip requirement is a necessary safety function for ensuring compliance with 10 CFR 50 criteria; although, the particular accident is extremely unlikely. I believe the consequences of tripping the pumps for non-LOCA events are sufficiently severe so as to take any measures necessary to preclude this action. Therefore, in the short term, I advocate changing the procedures to require the pump trip when pressure decreases to less than 1200 psig. In the long term, the feasibility of an automatic pump trip should be investigated.

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Enclosures:
1. Analysis Summary in Support of
   an Early RC Pump Trip
2. Supplemental Small Break Analysis
3. Duke Power Company, Incident Report B-175