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MEMORANDUM FOR: R. Vollmer, Director, TMI Support

THRU: G. R. Mazetis, Section Leader, Reactor Systems Branch, DSS

FROM: S. F. Newberry, Reactor Systems Branch, DSS

SUBJECT: TMI-1 RESTART REVIEW: EMERGENCY FEEDWATER SYSTEM CAPACITY AND ANTICIPATORY REACTOR TRIP

Two of the Bulletins and Orders Task Force requirements for TMI-1 and the other B&W reactors are the upgrade of the Emergency (Auxiliary) Feedwater System reliability and the installation of Anticipatory Reactor Trips on loss of feedwater and turbine trip. This memorandum is intended to document the status of our review of certain portions of these two areas on TMI-1 and to recommend respective staff positions. The discussion regarding Emergency Feedwater is specifically related to the sizing or capacity design basis of the system. The attached enclosures address each of these areas. We intend to proceed as indicated unless otherwise directed.

Scott F. Newberry
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Enclosures:
1. Emergency Feedwater Sizing Calculation
2. Anticipatory Reactor Trip on Loss of Feedwater
3. Oconee Loss of Feedwater Data
4. TMI-1 Loss of Feedwater Data
5. TMI-1 Reactor Trip Parameters

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Enclosure 1

Emergency Feedwater Sizing Calculation

Our review and evaluation of the TMI-1 EFW System must determine the bounding Emergency Feedwater System flow requirement for all transients and accidents (LOCA and non-LOCA) and the capability of the system to deliver the necessary water to the steam generators. The minimum acceptable requirement will appear in the TMI-1 Technical Specifications.

Our current understanding with Met-Ed is that they will provide the staff the following:

1. List of all events needing EFW to mitigate the consequences.
2. Justification that the bounding non-LOCA calculation will serve as a conservative basis for sizing the EFW system for non-LOCA core cooling considerations. In other words, they must show that the calculation they submit will bound all of the non-LOCA events requiring EFW.
3. The non-LOCA bounding event will be a loss of feedwater using FSAR type assumptions to maximize heat removal requirements (1.2 ANS decay heat, 2% power level measurement uncertainty, RCP heat input). The calculation will not take credit for "anticipatory reactor trip" since it will not occur under all conditions (see Enclosure 2).

The analytical method utilized will be CADDS.

The acceptance criteria for the event will be:

1. Reactor Coolant System pressure remains less than 110% of design pressure (2750 psig).
2. No fuel failure (DNBR >1.30).

This transient calculation would serve as the EFW sizing basis for all non-LOCA events (assuming our review finds no problems). It should be noted that for this conservative sizing calculation, the above acceptance criteria do not preclude lifting the PORV, although credit for pressure relief through the valve will not be assumed. The basis for small LOCA events would still be the generic calculations in reference 1 which assume 550 gpm with a 20 minute time delay for initiation.

Met-Ed expects to show that 500 gpm is adequate in their non-LOCA sizing calculation. Therefore, the flow requirement which would appear in the Technical Specifications is 550 gpm. The above approach to demonstrate adequate sizing of the EFW system is a reasonable basis for allowing restart of TMI-1.

Enclosure 2

Anticipatory Reactor Trip System (ARTS)

The ARTS were required by the Bulletins and Orders to help reduce the frequency of challenges to the PORV (in addition to lowering the high pressure trip setpoint and raising the PORV setpoint). Most of the current generic ART modifications generate a loss of main feedwater reactor trip on low main feed pump control oil pressure on both pumps (which is what trips the main feed pumps). It appears that the main feed pumps could be expected to trip on most of the loss of feedwater events that have occurred to date, including the TMI-2 accident, however, we have recognized that they would not necessarily trip for all possible loss of feedwater scenarios. For example, review of the Oconee loss of feedwater data (reference 2) shows that if the ARTS modification had been installed at the Oconee units, an ART would have been initiated for approximately 65% of the reported loss of feedwater events. A summary of the Oconee data is in Enclosure 3.

The change of the PORV and reactor trip setpoints (2450 psig and 2300 psig respectively) are intended to reduce the frequency of opening the PORV for anticipated transients. The bases for this approach are the best estimate calculations in reference 1 and the operating history of B&W plants since the changes were made (no PORV openings as a result of transients).

The ARTS were required to increase the margin attained by the revised setpoints (above). Based on the similarity of the Oconee units to TMI-1[1], the margin available even without ART, and the design, procedural and operator training modifications required to facilitate recognition and mitigation of a stuck open PORV (should it open), the proposed ART parameters of MFP control oil pressure and turbine trip are reasonable to meet the short-term Bulletin and Order requirements.

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1. This similarity is based on the assumption that Met-Ed installs main feedwater and condensate low suction pressure pump trips similar to those at the Oconee plants.

It is also interesting to note that of the two loss of feedwater events that have occurred in the operating history of TMI-1, it is not obvious that either event would have initiated on ART (reference 3). This is certainly true for the loss of instrument air event, but is not completely clear for the second event due to insufficient data. (see Enclosure 4)

In the longer term, we recommend that a more complete generic study be conducted for all B&W plants to evaluate the need for a more "encompassing" reactor trip which is diverse to the high pressure trip for loss of heat sink events. While the 177 FA plants do have a high reactor outlet temperature reactor trip, we note that ARTS is the only time that secondary system reactor protection parameters are utilized (see Enclosure 5). A promising direction perhaps is a reactor trip based upon mismatch of the primary side (power) with the secondary side (feedwater). Such a trip would probably cover all the Oconee loss of FW events, yet not compromise routine operations maneuvers (load changes, etc). Alternately, a more selective combination of secondary side parameters may provide a "better" coverage of pressure transients. An additional consideration is the current SRP requirement of section 5.2.2, Overpressurization Protection, which gives no credit for the first direct reactor trip which would be high pressure. B&W plants should be evaluated considering this requirement.

In summary, the safety-grade ART inputs proposed by the licensee are acceptable for TMI-1 restart; however, the need for a longer term generic study should be determined for all B&W plants concerning an alternate or additional diverse reactor trip.

Enclosure 3

Oconee 1, 2, 3 Data

Unit	Reactor Trips	Trip Due to LOFW	MFP Trip Prior to Reactor Trip
1	72	16	12
2	44	12	8
3	34	8	4
Total	150	36	24[1]

Notes: 1. 21 of 24 MFP trips above 10% reactor power (10% reactor power is the currently proposed bypass setpoint for the loss of feedwater anticipatory reactor trip)

Reactor Power (%)	No. of Events	MFP Trip Prior to Reactor Trip	Percent
0-10	3	3	100
11-20	10	6	60
21-50	12	10	83
51-75	5	4	80
76-100	6[1]	1 (possibly 2)	17

Notes: 1. 1 (possibly 2) of 6 would have generated ART; others caused by ICS malfunctions which "ran-back" main feedwater; therefore, no MFP trip or ART would have occurred.

Enclosure 4

TMI-1 Loss of Feedwater Data

Date	Power Level	Cause/Description
6/18/74	7%	Loss of Instrument Air - Caused main feed control valves to shut. Operator tripped reactor manually.
7/13/74	15%	High differential pressure across strainers on the suction side of C Condensate Booster Pump. Pump burned up due to low suction pressure. Reactor tripped on variable pressure/temperature reactor trip. (Status of main feed pumps is not clear.)

TABLE 2.3-1

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shut[illegible] Byp[illegible]
1. Nuclear power, Max. % of rated power	105.5	105.5	105.5	5.0(
2. Nuclear power based on flow(2) and imbalance, max. of rated power	1.08 times flow minus reduction due to imbalance(s)	1.08 times flow minus reduction due to imbalance(s)	1.08 times flow minus reduction due to imbalance(s)	Bypa[illegible]
3. Nuclear power based(5) on pump monitors, max. % of rated power	NA	NA	91%	Byp[illegible]
4. High reactor coolant system pressure, psig, max.	2355	2355	2355	172[illegible]
5. Low reactor coolant system pressure, psig, min.	1800	1800	1800	Byp[illegible]
6. Variable low reactor coolant system pressure, psig, min.	$(16.25T_{out} - 7756)^{(1)}$	$(16.25T_{out} - 7756)^{(1)}$	$(16.25T_{out} - 7756)^{(1)}$	Byp[illegible]
7. Reactor coolant temp. F., Max.	619	619	619	619
8. High Reactor Building pressure, psig, max.	4	4	4	[illegible]

(1) Tout is in degrees Fahrenheit (F)
(2) Reactor coolant system flow, %
(3) Administratively controlled reduction set only during reactor shutdown
(4) Automatically set when other segments of the RPS (as specified) are bypassed
(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.

References

1. Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plants, May 7, 1979.

2. Data on Oconee Loss of Feedwater Events - submitted informally to NRC staff by R. Wright (B&W), December 14, 1979.

3. Reactor Trip Summary for TMI-1 - submitted informally by Met-Ed during TMI-1 restart review as part of their review of operating experience in accordance with IE Bulletin 79-05A.