CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9609270295 DOC.DATE: 96/09/24 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244
AUTH.NAME AUTHOR AFFILIATION
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION
VISSING,G.S. Project Directorate I-1 (PD1-1) (Post 941001)

See Reports

SUBJECT: Informs that scope of of assessment includes review of pressure locking & thermal binding evaluations that have been previously performed.

DISTRIBUTION CODE: A056D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 8+155
TITLE: Generic Ltr 95-07 - Pressure Locking & Thermal Binding of Safety Rela

NOTES:License Exp date in accordance with 10CFR2,2.109(9/19/72). 05000244

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	NRR/DRPE/EATON	1	1	PD1-1 PD	1	1
	~~VISSING,G.~~	1	1			
INTERNAL:	FILE CENTER 01	1	1	NRR/DE/EMEB/B	1	1
EXTERNAL:	NOAC	1	1	NRC PDR	1	1
	NUDOCS ABSTRACT	1	1			

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D-5(EXT. 415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

MAE

TOTAL NUMBER OF COPIES REQUIRED: LTTR 8 ENCL 8

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649-0001 AREA CODE 716 546-2700

ROBERT C. MECREDY
Vice President
Nuclear Operations

September 24, 1996

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: Request for Additional Information - Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," R.E. Ginna Nuclear Power Plant (TAC No. M93466)
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

By letter dated June 18, 1996, a request was made by the NRC for RG&E to provide additional information in support of RG&E's 180 day response to NRC Generic Letter 95-07 which was submitted on February 16, 1996.

RG&E is presently conducting a MOV program self assessment utilizing industry resources to determine MOV program completeness. The scope of this assessment includes review of the pressure locking and thermal binding evaluations that have been previously performed. As such, the information provided herein reflects changes consistent with current industry practices that have been identified as part of this assessment. Any additional changes that may be identified by the continuing self assessment process that would affect the conclusions of any pressure locking or thermal binding information that is presented herein will be forwarded as it becomes available.

The following, based on current information, is submitted in response to your requests.

<u>Request 1.a</u>
The Licensee's submittal states that these valves *(857A,B,C)* could experience back leakage from the RCS during the post LOCA injection phase which could pressurize the RHR system to an estimated 614.7 psig. Please discuss whether these valves could experience back leakage from the RCS during normal power operation and become pressurized to a higher pressure.

Response
During normal operations, MOVs 852A and 852B (core deluge) and MOVs 720 and 721 (normal RHR inlet) would remain closed and therefore RCS back leakage and pressurization through the RHR system, and pressurization to 614.7 psig, is not postulated as it is for the post LOCA scenario. The RHR system is protected from over pressurization by a relief valve (203) with a setpoint of 606 psig. Back leakage from the RCS system through the SI system is postulated for normal operation. This pressure is limited by the SI pump suction relief valve set point of 210 psig (valve 1817). Therefore the postulated pressurization during normal operation is bounded by the pressurization during the post LOCA injection scenario.

Request 1.b
From a review of system diagrams, it appears that these valves *(857A,B,C)* may be potentially susceptible to thermally induced pressure locking from heat transfer from the RHR system during a design basis event. Please discuss your review of this potential susceptibility.

Response
MOVs 857A, 857B and 857C are not subject to thermally induced pressure locking due to heat transfer from the RHR system during a design-basis event due to the following:

- The inlets to the subject MOVs are separated by an extremely long run of 6 inch piping (approximately 59 feet) from the RHR heat exchanger outlet tees where the process fluid heat source would exist during RHR recirculation following a large-break loss of coolant accident (LOCA). Since a large dead leg exists to isolate these MOVs from the heat source and since the elevation of the MOVs is below that of the heat source, it is not possible for any heat transfer to occur over that distance.

- Under the analyzed case for RHR recirculation following a LOCA, the maximum temperature at the outlet of the RHR heat exchanger is approximately 180°F. This process fluid temperature is not sufficient to cause any appreciable heat transfer through 59 feet of adjacent dead leg piping.

Request 1.c
Please discuss your basis for using a friction coefficient of 0.5.

Response
The open friction coefficient of 0.5 results from an open valve factor of 0.5 and a disc seat angle of 0 degrees. The open friction coefficient of 0.5 for MOVs 857A, 857B and 857C was derived, as a bounding friction coefficient, from the results of

in-situ differential pressure testing of these MOVs. Based on a nominal seat diameter of 4 inches, MOV 857B had an open valve factor of 0.356 and MOV 857C had an open valve factor of 0.469. Although MOV 857A had an open valve factor of 1.18 based on the test data, this data is considered questionable due to anomalies identified during testing and since a torque-thrust cell was not employed during the test. Current methodology uses mean seat diameter (4.5 inches)and accounts for stem rejection force and torque reaction factor. The current methodology yields lower friction coefficients, thereby further enveloping the 0.5 value used in the calculation.

The use of a friction coefficient of 0.5 for all three MOVs is further justified based on the results of static testing of these MOVs performed during the 1996 outage. This testing demonstrated that the unseating thrust requirements have remained similar to previously determined requirements and were extremely low. That is, testing prior to 1994 indicated a maximum unseating thrust of 669 lbs. The most recent tests indicate the unseating thrust requirements range from 586 lbs to 770 lbs. Therefore, no indication of appreciable friction coefficient differences among the MOVs is evident nor has any significant friction coefficient increase over time occurred due to age-related degradation.

It is also noted that RG&E intends to perform additional testing on valve 857A using a torque-thrust cell to confirm an actual friction coefficient of less than 0.5.

<u>Request 1.d</u>
The licensee's calculation includes an unsealing load of 669 lbf. The NRC staff believes that, due to the wedging/unwedging mechanism of Anchor-Darling double disk gate valves, this load can be considerably higher. In addition, the measured unsealing load can be different for each valve stroke. Please address these issues and provide the basis for including an unsealing load of 669 lbf.

<u>Response</u>
The pressure locking and thermal binding study of record was completed in 1994 using MOV Program data available at that time. Prior to 1994, diagnostic test results of MOVs 857A, 857B and 857C indicated that the measured unsealing load was 669 lbs.

The results of diagnostic testing performed during the 1996 outage validates this input and establishes that unsealing loads for these MOVs have remained extremely low and relatively constant as follows:

MOV	Unsealing Thrust
857A	586 lbs.
857B	770 lbs.
857C	769 lbs.

RG&E has included the highest recorded value in the latest operability calculation, Attachment 1. This calculation demonstrates that the unsealing load based on this latest data will not increase the required thrust to a value higher than the available thrust.

Request 1.e
Please provide your actuator output capability calculations for these valves and address and provide justification for any deviations from the Limitorque guideline.

Response
Operability calculations based on in-situ motor/actuator capability testing are provided as Attachment 1.

Present RG&E actuator sizing methodology utilizes a calculational basis and in-situ test data to verify actuator capabilities for operability. The initial screening process compares the 89-10 program calculated available thrust, based on published actuator data, to the calculated required thrust. If it is determined that the calculated available thrust is less than the required thrust, field testing data is utilized to determine actual available thrust. The field testing consists of the performance of an "in-situ motor/actuator capability test."[1] This test measures actual thrust for different torque switch settings via a load cell mounted to the top of the valve stem. The testing is performed by incrementally increasing the actuator torque switch setting and measuring actual output thrust. The thrust measured is the thrust at torque switch trip and therefore any inertia developed after the trip is not erroneously included. While higher output thrust may actually be available, testing is discontinued when the required target thrust value has been

[1] It is noted that in previous discussions with the NRC, the in-situ motor/actuator capability test was erroneously referred to as a "partial stall test" by RG&E. The in-situ motor/actuator capability test does not in fact involve motor stall. The test is also not analogous to previous industry performed motor stall testing (dynamometer) where actuator output capabilities were extrapolated from the measured motor output values. The in-situ motor/actuator capability test performed is a direct measurement of the actuator's output at the valve stem in the direction required to overcome pressure locking (i.e. opening direction).

demonstrated. This thrust data is then adjusted via calculations to account for identified differences in valve/actuator operating characteristics (i.e. stem factors, degraded voltage). This adjustment is based on bounding values for the group of valves being qualified. For the 857 valves, the test data compares favorably with test data from the 860 valves (unadjusted values of 10,832 lbs. [857] versus 11,016 lbs.[860]) which have a similar configuration and thus demonstrates repeatability of results.

In regard to deviations from the Limitorque guidelines it is noted that the operability calculation is not based on the Limitorque actuator capacity methodology. The one factor that is common however to both the Limitorque methodology and the in-situ motor/actuator capability testing methodology is valve factor. RG&E may use a larger valve factor than the Limitorque recommended value based on the results of in-situ tests performed at Ginna or by other licensees or based on industry data which indicates that greater values are warranted. When a correctly assumed valve factor is utilized to establish the target differential-pressure thrust requirement and the in-situ motor/actuator capability test demonstrates that sufficient thrust is available above the required thrust, then operability is ensured.

Request 1.f
Please provide your weak link analysis and calculations for these valves.

Response
Weak link analysis and calculations are provided as Attachment 2.

Request 2
The licensee's submittal discusses the susceptibility of valves 852A/B, RHR to Reactor Vessel Deluge, to pressure locking and thermal binding and states that analyses of these conditions have been performed. Please provide these analyses for our review.

Response
The pressure locking/thermal binding susceptibility review for MOVs 852A and 852B is provided as Attachment 3.

Request 3
From a review of system diagrams, it appears that the following valves may be potentially susceptible to pressure locking: 720, 721, 878A, 878C. Please provide your pressure locking susceptibility evaluations for these valves, and include associated calculations completed for our review.

Response
The pressure locking susceptibility evaluations for MOVs 720 and 721 are provided as Attachment 4. However, these MOVs have no safety related function to open.

Since MOVs 878A and 878C are in their safety-related position (closed) with AC power removed, the valves cannot be inadvertently opened and since these valves are not required to be opened for any design-basis transient response (per Attachment 5), they were not evaluated for susceptibility to pressure locking.

Request 4
In attachment 1 to GL 95-07, the NRC staff requested that licensees include consideration of the potential for gate valves to undergo pressure locking or thermal binding during surveillance testing........ The staff stated that normally open, safety-related power-operated gate valves which are closed for test or surveillance but must return to the open position should be evaluated within the scope of GL 95-07. Please discuss if valves which meet this criterion were included in your review, and how potential pressure locking or thermal binding concerns were addressed.

Response
Valves (listed below) which are normally open and are closed for surveillance testing and are required to open have been reviewed and were found to be acceptable due to one or more of the following reasons:

1. The valve is not susceptible to pressure lock or thermal binding.

2. Technical Specifications are followed to ensure one train is available, i.e. the tested valve is returned to its normally open position prior to declaring it operable.

3. The valve actuator has sufficient available thrust to overcome pressure locking/thermal binding.

4. Hardware or procedure modifications have been performed to prevent pressure lock or thermal binding.

Valve Summary

515/516 These pressurizer power-operated relief valve block valves are double-disc gate valves modified (hole in disc) to preclude pressure locking.

704A/B When these residual heat removal pump suction valves are tested, Technical Specifications are followed to ensure one train remains operable.

813/814 These component cooling water isolation valves for the reactor support coolers have been evaluated as having sufficient available thrust to overcome pressure locking.

871A/B When these Safety Injection Pump C discharge valves are tested, Technical Specifications are followed to ensure the discharge path remains operable.

1815A/B When these series Safety Injection Pump C suction valves are tested, Technical Specifications are followed to ensure one safety injection pump suction path remains operable.

4615/4616 When these Auxiliary Building service water isolation valves are tested, Technical Specifications are followed to ensure one train remains operable.

Request 5

Through review of operational experience feedback, the staff is aware of instances where licensees have completed design or procedural modifications to preclude pressure locking or thermal binding which may have had an adverse impact on plant safety due to incomplete or incorrect evaluation of the potential effects of these modifications. Please describe evaluations and training for plant personnel that have been conducted for each design or procedural modification completed to address potential pressure locking or thermal binding concerns.

Response

All design changes are handled in accordance with the Plant Change Record (PCR) process. This process includes both a safety review/evaluation, as applicable, and a training plan. Example of this process is the modification of the 852A and 852B valves. PCR 96-085 modified the actuator thrust capabilities for these MOVs. As part of the modification process safety evaluation (SEV 1072) was performed. This safety evaluation addresses functional impacts as well as Technical Specification, UFSAR, Regulatory and Accident Mitigation issues. A training plan, in the form of a Training Notification Letter was also issued.

All procedure changes are handled in accordance with the Procedure Change Notice (PCN) process. This process includes a safety review/evaluation, as applicable, and the submittal of changes to training for review. Operations personnel are trained on these procedures in accordance with the operations personnel qualification program. No specific procedure changes to date have been issued to address PL/TB concerns.

Very truly yours,

Robert C. Mecredy

xc: Mr. Guy S. Vissing (Mail Stop 14C7)
Project Directorate I-1
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

US NRC Ginna Senior Resident Inspector