UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

December 8, 1999

MEMORANDUM TO: Elinor G. Adensam, Project Director
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

THRU: Lawrence T. Doerflein, Chief
Engineering Programs Branch
Division of Reactor Safety
Region I

FROM: A. Randolph Blough, Director
Division of Reactor Projects
Region I

R. V. Lloyd for

REFERENCES:

1) LER 50-244/99-003, Two Valves Declared Inoperable Results in Condition Prohibited by Technical Specifications (dated March 31, 1999)
2) RG&E's "Main Steam Non-Return Check Valve Closure Analysis" (dated May 27, 1999) 5/21/99?
3) Applicable excerpts from NRC Inspection Report 50-244/99-05 (dated August 6, 1999)
4) Letter from Robert C. Mecredy to USNRC, "NRC #40500 Team Inspection 50-244/99-05, dated 8/6/99" (dated August 23, 1999)
5) Letter from Robert C. Mecredy to USNRC, "Response to Questions Related to Main Steam Check Valve Performance per NRC Inspection 99-05" (dated September 24, 1999)

SUBJECT: PROPOSED TASK INTERFACE AGREEMENT (TIA) REGARDING THE ACCEPTABILITY OF CALCULATIONS USED TO DETERMINE THE OPERABILITY OF THE GINNA MAIN STEAM NON-RETURN CHECK VALVES (TAC NA7271)

Your assistance is requested to validate the technical adequacy of Rochester Gas and Electric's (the licensee) calculations used to support the current operability of the main steam non-return check valve (NRCVs) and to determine if the licensee has provided an adequate basis for demonstrating that operability.

During the shutdown for the 1999 refueling outage, the licensee tested the main steam NRCV as required by the plant's Technical Specifications. During the test, the licensee identified that the breakaway torque required to initiate closing of the NRCVs exceeded the acceptance criteria in the test procedure (900 ft-lbs and 600 ft-lbs, respectively). The licensee reported this event in Licensee Event Report (LER) 50-244/99-003 (Reference 1).

The licensee's actions regarding this event were reviewed during a subsequent NRC team inspection. The team determined that in 1992 the licensee tightened the packing on the NRCVs to address problems with packing leakage and main steam flow oscillations caused by check valve flutter. Also, in 1992, 1993, and 1999, the NRCV test procedure was revised to change the method the NRCVs were checked closed, to establish an acceptance criterion for the closing torque, and to modify the acceptance criterion, respectively. The team determined that for each change the licensee either did not perform a 10 CFR 50.59 evaluation or completed an inadequate evaluation in that the licensee failed to recognize the packing modification created an Unreviewed Safety Question (USQ).

The Updated Final Safety Analysis Report (UFSAR) Section 10.3.2.7 states that the main steam non-return check valves are free swinging gravity closing type check valves. Section 15.1.5.1 states that they, in conjunction with the main steam isolation valves, prevent blowdown of more than one steam generator in the event of a steam line rupture. The team determined that the packing modification increased the probability of occurrence of a malfunction of equipment previously evaluated in the safety analysis report and, as a result, introduced an USQ.

In response to the team's conclusion, the licensee provided complex engineering calculations (Reference 2) to demonstrate that the NRCVs remained operable. The licensee maintained that the calculations showed that there would be sufficient reverse steam flow to provide the breakaway torque needed to close the NRCVs for the spectrum of steam line break sizes below which containment pressure from the blowdown of both steam generators would be less than containment design pressure. The team identified several concerns regarding the licensee's calculations, and the licensee was requested to respond to a set of questions that was attached to the inspection report (Reference 3). Further discussions between the NRC and the licensee occurred regarding the NRCVs, and the licensee provided an initial response and summary of those discussions in a letter to the NRC dated August 23, 1999 (Reference 4).

In the response to the inspection report and attached questions regarding the NRCVs (Reference 5), the licensee maintained that the NRCVs were operable; in that, they would perform their safety function for the limiting steam line break, with substantial margin. The licensee indicated that this was confirmed by an independent assessment by Duke Engineering and Services. The licensee also indicated that the NRCV counterweights were moved to reduce the required breakaway torque by about 150 ft-lbs, which was not considered in the calculations and would provide even more margin. In the long term, the licensee indicated modifications would be pursued to return the NRCVs to a condition more representative of the original design (i.e., gravity closing). Notwithstanding, the licensee's calculations are complex and NRR assistance is needed to provide an adequate technical review.

This TIA has been discussed with Guy Vissing. The Region I point of contact is Lawrence T. Doerflein, of the Division of Reactor Safety, at (610)337-5378. Please complete this TIA by June 30, 2000.

Enclosures: References (1), (2), (3), (4), and (5)

Reference 1

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

March 31, 1999

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

Subject: LER 1999-003, Two Valves Declared Inoperable Results in Condition Prohibited by Technical Specifications
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

The attached Licensee Event Report LER 1999-003 is submitted in accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (i) (B), "Any operation or condition prohibited by the plant's Technical Specifications".

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Regional Administrator, Region I
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475 Allendale Road
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U.S. NRC Ginna Senior Resident Inspector

IE22

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a

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TITLE (4)

Two Valves Declared Inoperable Results in Condition Prohibited by Technical Specifications

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
03	01	1999	1999 --	003 --	00	03	31	1999	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	3
POWER LEVEL (10)	0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
John T. St. Martin - Technical Assistant	(716) 771-3641

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On March 1, 1999, at approximately 1707 EST, it was determined that the required torque to initiate valve disc closure for the two main steam non-return check valves was greater than the acceptance criteria specified in plant test procedures.

Immediate corrective action was to declare both valves inoperable and enter Technical Specification Limiting Condition for Operation 3.0.3. Following an evaluation of the test data by Nuclear Engineering Services, it was determined that the valves were operable. The plant exited Limiting Condition for Operation 3.0.3.

The underlying cause of the event was changes in the methodology and materials for packing these valves, which resulted in a greater than anticipated shaft breakaway torque.

Corrective action to prevent recurrence is outlined in Section V.B.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. PRE-EVENT PLANT CONDITIONS:

Since 1992, Performance Monitoring technicians have performed surveillance test procedure PT-2.10.15, "Main Steam Non-Return Check Valve Closure Verification", using the test methodology established by Nuclear Engineering Services (NES). Performance of test procedure PT-2.10.15 satisfies Ginna Station Improved Technical Specifications (ITS) Surveillance Requirement (SR) 3.7.2.2 and satisfies the requirements of Section XI of the ASME Code for these valves. The required torque to initiate valve disc closure (breakaway torque) for the main steam non-return check valves (CV-3518 and CV-3519) has consistently been measured significantly lower than the acceptance criteria specified within the test procedure (600 ft-lbs).

On March 1, 1999, the plant was in Mode 3, cooling down to Mode 4 for a scheduled refueling outage. Both main steam isolation valves (MSIVs) were closed. At approximately 1707 EST, Performance Monitoring technicians were performing procedure PT-2.10.15. The technicians were utilizing a calibrated torque wrench with a range of 0 to 600 ft-lbs, as they had in previous years. The technicians could not initiate valve disc closure (achieve breakaway torque), even at the full range of the torque wrench. They consulted with supervision, and initiated a plant ACTION Report to document the inability to achieve check valve disc movement up to 600 ft-lbs of torque.

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- o March 1, 1999, 1707 EST: Event Date and Time and Discovery Date and Time.
- o March 1, 1999, 1734 EST: Both main steam non-return check valves are declared inoperable.
- o March 1, 1999, 1930 EST: Engineering Technical Evaluation determines that both main steam non-return check valves are operable.
- o March 1, 1999, 2018 EST: The Plant enters Mode 4, where ITS LCO 3.7.2 is not applicable. ITS LCO 3.0.3 for the main steam non-return check valves is exited.

B. EVENT:

On March 1, 1999, the plant was in Mode 3, cooling down to Mode 4 for a scheduled refueling outage. Both main steam isolation valves (MSIVs) were closed, as specified by the Initial Conditions for test procedure PT-2.10.15. The Performance Monitoring technicians notified the Shift Supervisor of the failure of the main steam non-return check valves to meet the closure torque acceptance criteria of test procedure PT-2.10.15.

The Shift Supervisor reviewed ITS Limiting Condition for Operation (LCO) 3.7.2, notified the NES staff of the event, and requested an engineering technical evaluation. At approximately 1734 EST the Shift Supervisor declared both valves CV-3518 and CV-3519 inoperable based on exceeding the acceptance criteria of test procedure PT-2.10.15. As specified in ITS LCO Required Action 3.7.2.E.1, with "one or more valves inoperable in flowpath from each steam generator (SG)", immediate entry into ITS LCO 3.0.3 is required. The Shift Supervisor directed entry into ITS LCO 3.0.3 at this time.

Performance Monitoring technicians obtained a torque wrench of larger range and again attempted to achieve breakaway torque. At approximately 700 ft-lbs torque, the valve disc for CV-3518 started to close, and at approximately 900 ft-lbs torque the valve disc for CV-3519 started to close. These as-found breakaway torque values were provided to NES staff.

NES staff performed an engineering technical evaluation of this event. At approximately 1930 EST, NES staffhadreviewed an engineering analysis (Design Analysis DA-ME-92-147) that had been performed previously for these valves, and determined that the as-found breakaway torque was within the bounds of the analysis. This information was provided to the Shift Supervisor.

While the valves were now capable of being declared operable, the plant continued the planned cooldown and entered Mode 4 at approximately 2018 EST on March 1, 1999. In Mode 4, ITS LCO 3.7.2 is not applicable, and ITS LCO 3.0.3 was formally exited at this time.

The entry into ITS LCO 3.0.3 as a result of declaring both CV-3518 and CV-3519 inoperable is considered to be a condition prohibited by Technical Specifications. Entry into ITS LCO 3.0.3 for any reason or justification is considered reportable per the NRC guidance in NUREG-1022 Revision 1.

C. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

None

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

None

E. METHOD OF DISCOVERY:

This event was discovered by Performance Monitoring technicians who were performing a routine surveillance test during the plant cooldown.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

F. OPERATOR ACTION:

The Shift Supervisor reviewed ITS LCO 3.7.2 and declared both valves CV-3518 and CV-3519 inoperable based on exceeding the acceptance criteria of test procedure PT-2.10.15. The Shift Supervisor directed entry into ITS LCO 3.0.3 at this time. The Shift Supervisor notified NES staff of the event, and requested an engineering technical evaluation. The operators continued the process of performing a plant cooldown per operating procedure O-2.2, "Plant Shutdown from Hot Shutdown to Cold Conditions".

After the plant was in Mode 4, ITS LCO 3.7.2 was not applicable and LCO 3.0.3 was exited for the main steam non-return check valves.

G. SAFETY SYSTEM RESPONSES:

None

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the condition prohibited by Technical Specifications was entering ITS LCO Required Action 3.7.2.E.1 for two valves inoperable, which required immediate entry into ITS LCO 3.0.3.

B. INTERMEDIATE CAUSE:

The intermediate cause of entry into ITS LCO 3.7.2.E.1 was the decision to declare both main steam non-return check valves inoperable for exceeding the acceptance criteria of Steps 6.1.3 and 6.2.3 of test procedure PT-2.10.15.

C. ROOT CAUSE:

The underlying cause for exceeding the acceptance criteria was changes in the methodology and materials for packing these valves, instituted during the previous outage. These changes in methodology and vendor-recommended replacement shaft bushing materials were made in order to provide improved shaft sealability and vibration mitigation, and resulted in a greater than anticipated shaft breakaway torque. Over time, during the previous plant operating cycle, heat and moisture were absorbed by the packing, which caused the shaft friction to increase to the as-found values of 700 and 900 ft-lbs, which were higher than anticipated, based on testing results from previous years.

IV. ANALYSIS OF EVENT:

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (i) (B), "Any operation or condition prohibited by the plant's Technical Specifications". Declaring both main steam non-return check valves inoperable resulted in entry into ITS LCO 3.0.3. Since the plant entered ITS LCO 3.0.3, this condition is reportable.

An assessment was performed considering both the safety consequences and implications of this event with the following results and conclusions:

There were no operational or safety consequences attributed to not meeting the acceptance criteria specified in procedure PT-2.10.15 because:

- o The acceptance criteria in test procedure PT-2.10.15 was conservatively chosen in 1992 to be well below the value calculated in Design Analysis DA-ME-92-147. This conservative value had been utilized as the acceptance criteria in test procedure PT-2.10.15, prior to defining the operability requirements in ITS SR 3.7.2.2. The engineering technical evaluation performed on March 1, 1999, determined that the as-found breakaway torque values for the non-return check valves were within this previous analysis.

- o The two MSIVs isolate steam flow from the secondary side of the steam generators (SGs) following a Design Basis Accident (DBA). Both MSIVs were closed, as specified in the Initial Conditions of test procedure PT-2.10.15, prior to initiation of the surveillance test on March 1, 1999. The MSIVs are designed to work with the main steam non-return check valves, located immediately downstream of each MSIV, to preclude the blowdown of more than one SG following a steam line break (SLB).

Based on the above, it can be concluded that the public's health and safety was assured at all times.

V. CORRECTIVE ACTION:

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

Immediate corrective action was to declare both valves inoperable and enter ITS LCO 3.0.3. Following an evaluation of the test data by NES, it was determined that the valves were operable. The plant exited ITS LCO 3.0.3.

The plant is still in the 1999 refueling outage.

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TEXT CONTINUATION

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- o Packing gland torque for these check valves will be adjusted to a value specified by the IST Engineer. An as-found baseline breakaway torque value will be obtained for each valve during the 1999 outage.

- o The design analysis will be revised to provide acceptance criteria, both for the ASME Code degradation value and for determination of valve operability.

- o A "reference value" will be established in accordance with ASME/ANSI OM-1987 Part 10 for breakaway torque for these valves. This value will be included in a future revision to test procedure PT-2.10.15.

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

None

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: No documentation of similar LER events with the same root cause at Ginna Nuclear Power Plant could be identified.

C. SPECIAL COMMENTS:

None

Post-it Fax Note 7671 | Date 6/11 | # of pages 11
To BOB SCHIN
Fax 404 562-4983
From TOM HARDING
Phone # 716-771-3384

Main Steam Non Return Check Valve Closure Analysis

Ginna Station

Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

DA-ME-92-147

Revision 2

5/27/99
Approval Date

Prepared by: KAMull
Assigned Engineer
5/21/99
Date

Reviewed by: James F. Durand
Independent Reviewer
5-21-99
Date

REVISION STATUS SHEET

Revision Number	Affected Sections	Description of Revision
0	All	Original Issue
1	All	Revised due to new steam conditions resulting from steam generator replacement
2		Revised due to re-analysis of most limiting accident conditions

1.0 **Purpose**

1.1 The purpose of this design analysis is to evaluate the closing moment applied to the valve disc of the main steam non-return check valve with its disc stuck completely open on the back stop under conservatively-assumed steam flow conditions which envelope current and future potential variations in plant conditions.

1.2 The net closing moment shall consider the effects of steam flow, weights and positions of the disc assembly and counterweights and fluid mechanics consideration of pressure variation in a flowing fluid.

1.3 The resultant calculated moment shall be compared to as-found torque values measured during manual closure activities.

1.4 Recommended methodology for periodic inservice testing shall be based on the results of this design analysis.

1.5 Revision 1 to this design analysis has been prepared to incorporate revised steam flow conditions which have resulted from the replacement of the original steam generators.

1.6 Revision 2 to this design analysis has been prepared to incorporate the results of re-analysis determining the most limiting steam line break case.

2.0 **Conclusions**

2.1 It is concluded that a sufficient moment will be present to ensure closure of check valves 3518 and 3519 under the conditions described in this analysis and that CATS item CO 2163 may be closed.

2.2 The closing moment present under the Revision 1 steam flow conditions is sufficient to ensure closure of check valves 3518 and 3519 under the conditions described in this analysis.

2.3 The closing moment present under the Revision 2 steam flow conditions is sufficient to ensure closure of check valves 3518 and 3519 under the conditions described in this analysis. Attachment 1 is a spreadsheet that has been developed and verified to be in agreement with the calculation methodology developed within this analysis in order to represent the varying parameters and the corresponding reverse steam flow closing moment.

3.0 <u>Design Inputs</u>

3.1 R. E. Ginna Nuclear Power Plant Updated Final Safety Analysis Report (UFSAR), Section 10.3.2.2, Revision 14.

3.2 The distance from the centerline of the counterweights to the centerline of the valve shaft is 12 inches when the counterweights are fully retracted and 21 inches when the counterweights are fully extended as measured in the field. Full extension is used in Section 7.10 as a conservative input.

3.3 The weight of each counterweight is 150 lb. (Ref 4.5).

4.0 <u>Referenced Documents</u>

4.1 DA-NS-99-054, "Main Steam Non-Return Check Valve Flow During A Small Steam Line Break", Revision 0.

4.2 ASME Steam Tables, Fourth Edition.

4.3 Engineering Fluid Mechanics, Second Edition, Roberson/Crowe.

4.4 Instruction Manual, Main Steam Isolation and Main Steam Check Valves Manufactured By: Atwood & Morrill Co., Salem, MA, RG&E Vendor Manual No. A585-0186.00.

4.5 Atwood & Morrill Co. Drawing 20729-H, 30 Inch O.D. Pipe Main Steam Isolation Check Valve.

4.6 OMa-1988, Part 10, Inservice Testing of Valves in Light-Water Reactor Power Plants.

5.0 <u>Assumptions</u>

5.1 Saturated steam flow is assumed to be non-compressible since pressure is relatively constant across the valve.

5.2 The projected area "A_P" used to calculate the closing moment was reduced by 0.5 at a disc position of 75° from vertical. This assumes that 50% of the disc is out of the flow stream at this disc position. This is conservative since the approximately 3 inch long back stop ensures the valve disc will be exposed to the reverse flow stream and the disc will divert the steam flow such that the whole area will be in the flow stream.

5.3 Check valve disc is assumed to be a flat circular disc.

6.0 **<u>Computer Codes</u>**

6.1 Excel Spreadsheet (Attachment 1) - Verfied by comparison with Section 7 Results.

7.0 <u>Analysis</u>

7.1 Conditions:

<u>Note</u>: Reference 4.1 supplies system parameters and multiplication factors for nominal steam flow that occur at specific times during the analyzed accident. For this analysis, the conditions at time T = 1.0 seconds were used since this is the data point where check valve closure is assumed to occur.

Pressure (P) = 800 psia (saturated) [from Ref. 4.1]

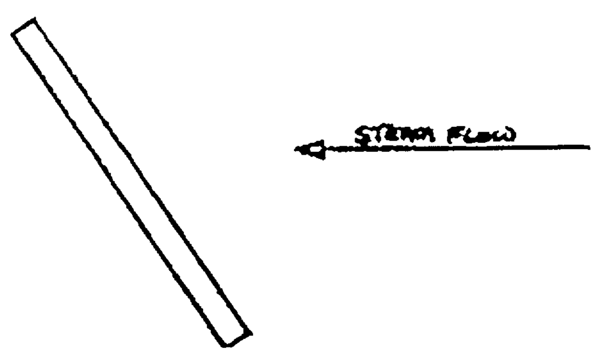
Temperature (T) = 518.2°F [from Ref. 4.2]

Specific Volume (V) = 0.5689 ft^3/lb_m [from Ref. 4.2]

Density (ρ) = 1 / V = 1.75 lb_m/ft^3

SLB Mass Flow Rate (m) = 603.3 lb_m/sec [from Ref. 4.1]

7.2 Treating the check valve disc as a flat circular disc:



The force F_D acting to close the disc is the total drag force and is calculated by:

$$F_D = C_D \rho (v^2/2) A_P$$ [from Ref 4.3]

Where: C_D = Drag Coefficient
ρ = Fluid Density
v = Fluid Velocity
A_P = Projected Area of Disc Perpendicular to Direction of Flow

Note: C_D is a function of impact angle and is maximized when disc is perpendicular to flow. Since Reference 4.3 only provides values of C_D for perpendicular applications, the projected area was used to compensate for not varying C_D.

7.3 Determine Value of C_D:

7.3.1 Calculate Fluid Velocity (v):

$$v = m / \rho A$$ [from Ref. 4.3]

Where: m = Mass Flow Rate
ρ = Fluid Density
A = Area of 30 inch, 1.25 inch nominal wall pipe from Line Specification 600-1.

$$A = \pi d^2 / 4$$

$$= \pi (27.5)^2 / 4 = 593.9 \text{ in}^2$$

$$v = \frac{603.3 \text{ lb}_m/\text{sec}}{(1.75 \text{ lb}_m/\text{ft}^3)(593.9 \text{ in}^2)(\text{ft}^2/144 \text{ in}^2)}$$

$$v = 83.6 \text{ ft/sec}$$

7.3.2 Calculate Reynolds Number (N_R):

$$N_R = v D / \nu$$ [from Ref. 4.3]

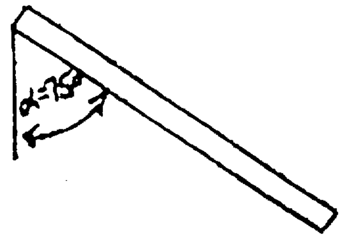
Where: v = Fluid Velocity
D = Disc Diameter = 25.5 in.
ν = Kinematic Viscosity = 6.0×10^{-6} ft²/sec from Reference 4.2, Figure 8

$$N_R = \frac{(83.6 \text{ ft/sec})(25.5 \text{ in}/12 \text{ ft})}{6.0 \times 10^{-6} \text{ ft}^2/\text{sec}}$$

$$= 2.9 \times 10^7$$

From Reference 4.3, Table 11-1, for $N_R > 10^4$ for a disc, $C_D = 1.17$

7.4 Calculate the Projected Area of the Disc (A_P):



(from Reference 4.5 check valve full open)

A_P = Area of Disc ($\cos\alpha$)

$$= \frac{\pi D^2 \cos\alpha}{4}$$

$$= \frac{\pi (25.5 \text{ in})^2 (\cos 75°)}{4}$$

$= 132.18 \text{ in}^2$

7.5 Calculate Drag Force (F_D) for $\alpha = 75°$:

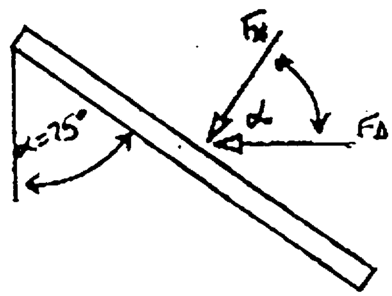
Note: Approximately one-half of the disc surface is above the flow stream in the capped region of the valve body. Therefore, only one-half of the projected area of the disc (A_P) will be used to calculate the closing force.

$$F_D = C_D \rho (v^2/2) (A_P/2)$$

$$=(1.17)(1.75 \text{ lb}_m/\text{ft}^3)\left(\frac{83.6^2 \text{ ft}^2/\text{sec}^2}{2}\right)\left(\frac{132.18 \text{ in}^2}{2}\right)\left(\frac{1 \text{ ft}^2}{144 \text{ in}^2}\right)\left(\frac{1 \text{ lb}_f \text{ sec}^2}{32.2 \text{ lb}_m \text{ ft}}\right)$$

$= 102 \text{ lb}_f$

7.6 Calculate the normal force acting on the check valve disc:



$$F_N = F_D \cos\alpha$$

$F_N = (102\ lb_f)\cos 75°$

$= 26.3\ lb_f$

7.7 Calculate the moment due to steam flow acting to close the disc:

Distance from hinge pin to centerline of disc = 15.5 in [from Ref.4.4]

Due to disc geometry, F_N acts at approximately ¼ D from top of disc,

Total Moment Arm = 15.5 in + [¼(25.5)]

= 21.875 in

$Moment_{Flow} = F_N$ (21.875 in / 12 in/ft)

= (26.3 lb_f) (21.875 in / 12 in/ft)

$Moment_{Flow}$ = 47.9 ft-lb = **M-DRAG** (Attachment 1)

7.8 Calculate the moment due to disc and disc arm assembly weight acting to close the disc:

Distance from hinge pin to centerline of disc = 15.5 in

Disc Angle of 75°

Weight of the assembly = 725 lbs [from Ref. 4.4]

$Moment_{DA}$ = (725 lb)(15.5 in)(1 ft/12 in)(sin 75°)

= 904.5 ft-lb = **M-DISC** (Attachment 1)

7.9 Calculate the moment due to pressure variation in a flowing fluid as represented by the Bernoulli equation:

<u>Notes</u>: 1. For this calculation, the valve disc is conservatively assumed to be completely out of the flow stream.

2. Due to the geometry of the check valve, the pressure above the valve disc is greater than the pressure acting below the disc. The pressure above the disc approached the stagnation pressure of the fluid since it exists in an

area of little or no flow. The pressure below the disc is the static pressure of the fluid as it passes the disc. Due to the frictional losses associated with the disc configuration, the static pressure of the fluid downstream of the valve should be lower than the static pressure upstream of the valve. To approximate these two effects, the differential pressure across the disc is assumed to be the velocity head of the upstream fluid, therefore:

$$\Delta P = \frac{\rho v^2}{2 g_c}$$

$$= \frac{(1.75 \text{ lb}_m/\text{ft}^3)(83.6 \text{ ft/sec})^2}{2 (32.2 \text{ lb}_m\text{-ft/lb}_f\text{-sec}^2)(144 \text{ in}^2/\text{ft}^2)}$$

$$= 1.31 \text{ psid}$$

Applying this pressure differential over the area of the valve disc results in a moment of:

$$\text{Moment}_{\Delta P} = \Delta P \text{ A } (15.5 \text{ in})(1\text{ft}/12 \text{ in})$$

$$= (1.31 \text{ lb}_f/\text{in}^2)(510.7 \text{ in}^2)(15.5 \text{ in})(1\text{ft}/12 \text{ in})$$

$$= 864.1 \text{ ft-lb} \qquad = \qquad \text{M-STAG (Attachment 1)}$$

7.10 Calculate the effects of the counterweights (opposite moment introduced at 21 inches from shaft centerline at an angle of 75°):

$$\text{Moment}_{Cwt} = (2 \text{ weights})(150 \text{ lb/wt})(21 \text{ in})\frac{(1\text{ft})}{12\text{in}}(\sin 75°)$$

$$= 507.11 \text{ ft-lb} \qquad = \qquad \text{M-CW (Attachment 1)}$$

7.11 Since the moments associated with the weights of the disc arm and counterweight are present under breakaway torque testing and will be present when design-basis closure is required, the total closing moment that will be expected to initiate closure of these check valves is the sum of the moments due to flow and pressure:

$$\text{Moment}_{TOT} = \text{Moment}_{Flow} + \text{Moment}_{\Delta P}$$

$$\text{Moment}_{TOT} = 47.9 \text{ ft-lb} + 864.1 \text{ ft-lb}$$

$$\text{Moment}_{TOT} = 912 \text{ ft-lb} \qquad = \qquad \text{M-TOT (Attachment 1)}$$

8.0 Results

8.1 The result of this analysis is that the total closing moment for the main steam non-return check valves has been calculated to be 912 ft-lb under the accident conditions provided. This closing moment neglects both the moment due to the counterweight, which would tend to keep the check valve open, and the moment due to the disc and disc arm, which tends to help the check valve close.

8.2 Data trends from PT-2.10.15 indicate that changes in check valve shaft packing installation methodology and materials greatly affects the amount of shaft breakaway torque measured during testing. The largest breakaway torque value measured during PT-2.10.15 was 900 ft-lb. This analysis demonstrates that safety function closure of these check valves is ensured due to the total closing moment that would be available under design-basis steam flow conditions.

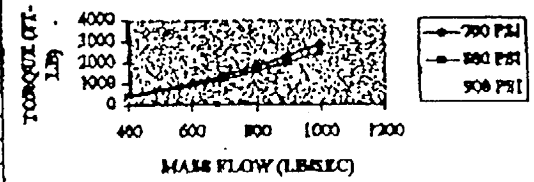
8.3 Engineering continues to recommend that a reference breakaway torque reference value of 600 ft-lbs continues to be utilized for check valve shaft breakaway torque testing until a management decision is made regarding modification of these check valves to replace the packing stuffing boxes with end bushings. Valve Packing Improvement Program requirements for these check valves have been amended to provide a 600 ft-lb shaft breakaway torque target to be met during valve repacking activities. Acceptance criteria should be based on ASME/ANSI OMa-1988, Part 10, Section 4.3.2.4(b) in that the breakaway shall not vary by more than 50% from the established reference value.

8.4 A spreadsheet has been included as Attachment 1 which provides closing moments for a series of steam conditions. Case 3 represents the conditions analyzed herein and the verification of the Case 3 calculations as compared to the calculations performed within the body of this analysis validate the calculations performed within the spreadsheet for all cases.

CASE		1	2	3	4	5	6	7
W	LB/SEC	400	500	603.3	700	800	900	1000
P	PSIA	100	800	800	800	800	800	800
RHO	#/CU FT	1.7609	1.7609	1.7609	1.7609	1.7609	1.7609	1.7609
D-PIPE	IN	27.5	27.5	27.5	27.5	27.5	27.5	27.5
A-PIPE	SQ IN	593.96	593.96	593.96	593.96	593.96	593.96	593.96
V	FT/SEC	55.07	68.84	83.06	96.37	110.14	123.91	137.69
CD		1.17	1.17	1.17	1.17	1.17	1.17	1.17
ANGLE	DEG	75	75	75	75	75	75	75
D-DISC	IN	25.5	25.5	25.5	25.5	25.5	25.5	25.5
A-DISC	SQ IN	510.71	510.71	510.71	510.71	510.71	510.71	510.71
DP-DRAG	PSI	0.175	0.273	0.397	0.535	0.698	0.884	1.091
X-DRAG		0.5	0.5	0.5	0.5	0.5	0.5	0.5
F-DRAG	LBF	44.57	69.64	101.37	136.49	178.27	225.62	278.55
L-DRAG	IN	21.875	21.875	21.875	21.875	21.875	21.875	21.875
M-DRAG	FT-LB	21.83	32.45	47.83	64.40	84.11	106.45	131.42
DP-STAG	PSI	0.576	0.901	1.311	1.765	2.305	2.918	3.602
F-STAG	LBF	294.35	459.92	669.59	901.45	1177.40	1490.15	1839.69
L-STAG	IN	15.5	15.5	15.5	15.5	15.5	15.5	15.5
M-STAG	FT-LB	[illegible]	[illegible]	864.89	1164.37	[illegible]	1924.78	2376.27
M-TOT	FT-LB	[illegible]	[illegible]	912.73	1228.77	[illegible]	2031.23	2507.69
W-DISC	LB	725	725	725	725	725	725	725
L-DISC	IN	15.5	15.5	15.5	15.5	15.5	15.5	15.5
M-DISC	FT-LB	904.58	904.55	904.55	904.55	904.55	904.55	904.55
W-CW	LB	300	300	300	300	300	300	300
L-CW	IN	21	21	21	21	21	21	21
M-CW	FT-LB	507.11	507.11	507.11	507.11	507.11	507.11	507.11
M-FREE	FT-LB	397.44	397.44	397.44	397.44	397.44	397.44	397.44
M-NET	FT-LB	793.67	1024.56	1310.56	1626.28	2082.36	2428.66	2905.12

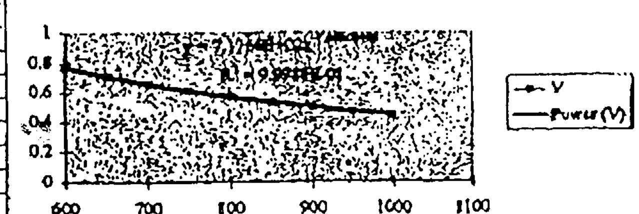
TORQUE			
FLOW	700 PSI	800 PSI	900 PSI
400	462.36	401.23	353.79
500	723.06	626.92	552.79
603.3	1052.68	912.72	804.8
700	1417.19	1228.77	1083.47
800	1851.02	1604.92	1415.15
900	2342.7	2031.22	1791.05
1000	2892.22	2507.69	2211.17

FLOW TORQUE vs MASS FLOW RATE



P	V		RHO	RHO-C	DIFF
600	0.76975	0.772222	1.2991	1.294965	-0.32%
650	0.70843	0.708928	1.4116	1.410581	-0.07%
700	0.65556	0.654962	1.5254	1.526807	0.09%
750	0.60949	0.60842	1.6407	1.643602	0.18%
800	0.56896	0.567811	1.7576	1.760932	0.19%
850	0.53302	0.532365	1.8761	1.878765	0.14%
900	0.50091	0.508033	1.9964	1.997073	0.04%
950	0.47205	0.472627	2.1184	2.115832	-0.12%
1000	0.44598	0.447023	2.2424	2.235019	-0.33%

STEAM PROPERTY vs PRESSURE



Design Analysis

DA-M6-92-147

Revision 0

Date 5/21/92

TOTAL P.11

AUGUST 6, 1999

EA 99-161

Dr. Robert C. Mecredy
Vice President, Ginna Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

SUBJECT: NRC 40500 TEAM INSPECTION 50-244/99-05 (CORRECTIVE ACTION PROGRAM EFFECTIVENESS)

Dear Dr. Mecredy:

This letter transmits the results of the NRC team inspection involving the review of the implementation of the corrective action program at the Ginna Nuclear Power Plant. The inspection was performed onsite from May 10-14 and May 24-28, 1999, using NRC Inspection Procedure 40500, "Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems." At the conclusion of the inspection, the findings were discussed onsite with Mr. Paul Wilkins, Senior Vice President, Mr. J. Widay, Plant Manager, and other members of the plant staff on May 28, 1999; and by telephone with Mr. Wrobel on June 24, 1999.

Overall, the team noted generally good implementation of the corrective action program. The problems were identified at a low threshold, the problem documentation and root cause determinations were satisfactory, corrective actions were developed and implemented in a timely manner, and management involvement was evident. Notwithstanding, continued emphasis is needed in the root cause evaluations for human performance errors. Of specific concern is the failure of your staff to pursue excessive overtime as a potential root cause for a reactor trip. This weakness in the control of overtime remains unresolved pending further review by the NRC

In addition, an apparent violation was identified involving inadequate safety evaluations for changes to the main steam non-return check valves. The changes to these check valves in 1992, 1993, and 1999 increased the probability of occurrence of a malfunction of equipment important to safety, and the changes introduced an Unreviewed Safety Question without obtaining the required NRC review and approval. This apparent violation is still under review, and additional information is needed regarding your position that the main steam non-return check valves currently meet their specified functional and acceptance criteria.

You are requested to respond with this additional information within 30 days of the receipt of this letter. You will be advised by a separate correspondence of the results of our deliberations on this matter. In addition, please be advised that the characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In accordance with 10CFR 2.790 of the NRC's "Rule of practice," a copy of this letter and enclosures will be placed in the NRC Public Document Room(PDR).

Sincerely

ORIGINAL SIGNED BY:

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 50-244
License No. DRP-18

Enclosure: Inspection Report No. 50-244/99-05

cc w/encl:
P. Wilkens, Senior Vice President, Generation
Central Records (7 copies)
P. Eddy, Electric Division, Department of Public Service, State of New York
C. Donaldson, Esquire, State of New York, Department of Law
N. Reynolds, Esquire
F. William Valentino, President, New York State Energy Research and Development Authority
J. Spath, Program Director, New York State Energy Research and Development Authority

c. Conclusion

RG&E's root cause determinations were generally satisfactory. Increased emphasis on improving the human performance evaluation portion of the root cause determination was noted. However, the effectiveness of this effort was not yet apparent, as plant events directly attributed to personnel error continued to occur. In addition, weaknesses in licensee evaluation of an excessive overtime issue were observed. The team also noted several examples of problems, not specifically related to human performance issues, which were not fully analyzed or evaluated during the root cause determination process to fully assess all contributing factors.

III. Engineering

E7 Quality Assurance in Engineering Activities

E7.1 Operability Determinations

a. Inspection Scope (40500)

The inspectors reviewed the licensee's guidance for performing operability determinations and reviewed 22 operability determinations that had been performed in 1999. The team identified several deficiencies regarding one operability determination for the main steam non return check valves.

b. Observations and Findings

Main Steam Line Non-Return Check Valves

Background

On March 1, 1999, the plant was in hot shutdown, cooling down for a scheduled refueling outage. During the performance of surveillance test PT-2.10.15, "Main Steam Non-Return Check Valve Closure Verification," the licensee identified that the torque required to initiate valve movement (breakaway torque) was in excess of the 600 ft-lbs of torque acceptance criteria. The operators appropriately entered Technical Specification 3.0.3, initiated an Engineering Technical Evaluation to evaluate operability, and continued with the plant cooldown. Prior to reaching cold shutdown, the Engineering Technical Evaluation determined that the main steam non-return check valves were operable provided the breakaway torque was less than 900 ft-lbs. The licensee reported this condition to the NRC in Licensee Event Report (LER) 99-003. The team reviewed the operablity determination associated with this condition.

Valve Maintenance History

The packing on the main steam check valves was tightened in 1992 to address problems with packing leakage and check valve flutter. The tighter packing changed the valves from free swinging gravity closing to valves that required approximately 600 ft-lbs of torque to close. During the 1997 refueling outage, the licensee repacked the main steam non-return check valves and left them with the required closing torque of less than 600 ft-lbs. During the plant operating cycle, the torque required to close the valves increased from 600 ft-lbs to a maximum of 900 ft-lbs. During the 1999 refueling outage, the main steam non-return check valves were again repacked and left with a required closing torque of less than 600 ft-lbs. On April 23, 1999, approximately one month after restart, the closing torque had increased to approximately 775 ft-lbs. The plant was restarted with the valves left in this condition.

Licensing Basis

The team reviewed the Updated Final Safety Evaluation Report (UFSAR) description of the main steam non-return check valves. The UFSAR stated in Section 10.3.2.7: "the main steam non-return check valves . . . are free swinging gravity closing type check valves. The check valves protect the main steam header against reverse flow from one generator to another in the event of a steam line rupture." The UFSAR, Section 15.1.5.1 states that: "Each steam line has a fast-closing MSIV and a non-return check valve. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, for a break upstream of the main steam isolation valve in one line, closure of either the non-return check valve in that line or the MSIV in the other line will prevent blowdown of the other steam generator."

Testing Procedures

The team reviewed PCN 93-4130, which processed PT-2.10.15, Rev. 2, dated March 12, 1993. This procedure revision incorporated the acceptance criterion of less than or equal to 600 ft-lbs This procedure revision included no safety evaluation. The stated basis for exclusion from a full safety evaluation was: "This change incorporates the change in test methodology recommended by DA-ME-92-024. This new method will provide a much greater degree of assurance that the subject valves are operable and will be capable of closing during all conditions of operation. This change does not place equipment in a configuration that is adverse to plant safety. This new test method is in full compliance with the code commitments of the Ginna Pump & Valve and In-Service Testing (IST) Programs." The team noted that the basis for exclusion from a full safety evaluation failed to recognize that this change represented a change to the plant as described in the UFSAR, and that a full safety evaluation was required.

The team additionally reviewed the current revision of PT-2.10.15 (Rev. 6). It included an acceptance criterion of 600 ft-lbs plus or minus 300 ft-lbs The team noted that it allowed a higher torque than Revs. 1 and 2, and thus further increased the probability that a main steam non-return check valve would fail to close during a main steam line break. Rev. 6, which had increased the maximum acceptable breakaway torque from

600 ft-lbs to 900 ft-lbs, had been processed and approved under PCN 99-4171, dated April 23, 1999. PCN 99-4171 included no safety evaluation. The stated reason for not including a 50.59 safety evaluation was that the change had been previously reviewed as design analysis DA-ME-92-147, Rev. 1, dated April 15, 1999. The team noted that the design analysis also had no safety evaluation.

Technical Evaluation

On March 1, 1999 an Engineering Technical Evaluation concluded that, with a measured closing torque of 900 ft-lbs, the main steam non-return check valves remained operable. This conclusion was based on "Design Analysis DA-ME-92-147, Rev. 0, dated November 10, 1992. The Design Analysis determined that at least 1567 ft-lbs torque was available to close the main steam non return check valves assuming ½ the design basis accident steam flow. Therefore, the licensee concluded that there was a significant margin above the maximum measured breakaway torque of 900 ft-lbs. The team identified a mathematical error in the Design Analysis that reduced the calculated closing torque to 963 ft-lbs. The Engineering Manager initiated action to correct the error in the calculation. Based on this Engineering Technical Evaluation, the licensee concluded that the main steam non-return check valves were operable and were not in a nonconforming condition.

The NRC conducted a detailed review of calculation DA-ME-92-147. The NRC review observed that the steam flow past the check valve, with flow reversal occurring at the time of the incident, presents a very complicated flow geometry and that a detailed flow field and pressure distribution on the valve disc is needed to properly analyze the effects on the check valve. Additionally, the licensee did not show that the uncertainty in the calculation was less than the available margin of torque needed to close the valve and it was not clear that the worst case condition was used regarding steam line break size and associated flow past the check valve. The team noted that, although the 963 ft-lbs available torque exceeded the measured 900 ft-lbs torque, it may not provide significant margin during a design basis double-ended main steam line break. Factors that influence the applicability and conservatism of the calculation for which additional information is needed include: basis for analysis method used (a 3-dimensional computational fluid dynamics code may be needed to properly model this complex flow condition); how the analysis method is validated; basis for assuming that the steam flow is non compressible since flow in the line is changing in mass flow rate and reversing direction; affects of a steam flow from steam generator that is blowing down until the check valve closes; and, basis for the check valve disc being treated as a flat circular disc when there is flow on both sides of the disc and when there are flow obstructions on the top of the disc.

Further, during a smaller main steam line break concurrent with the failure of a MSIV, the main steam non-return check valve may not close at all and may allow blowdown of both steam generators. This would represent an unanalyzed condition for steam generator integrity (both steam generators faulted), containment integrity (blowdown of both steam generators), steam generator tube integrity (emergency operating procedures [EOPs] would require the use of a faulted steam generator), reactor reactivity (potentially increased cooldown), and reactor vessel integrity (increased cooldown could overstress the reactor vessel).

The risk associated with this issue represents a minimal reduction in the margin of safety. For this event to be of concern, the main steam/feedwater system(s) within containment must be breached and a main steam isolation valve would need to fail to close. The probability of this sequence of events occurring is low. In addition, large dry containment buildings have been demonstrated to withstand internal pressure in excess of the design limits.

Corrective Actions

The team concluded that the main steam non-return check valves were in a non-conforming condition and that the licensee had not fully demonstrated operability. In response to the team's concerns, the licensee initiated AR 99-0890. As compensatory actions for the nonconforming condition, the licensee: 1) Submitted a Work Order to lower the position of the counterweights on 3518 and 3519 to reduce required break-away torque, 2) Initiated an evaluation of removing the counterweight assembly and arms to further reduce the required break-away torque, 3) Initiated an evaluation of modifying the check valves to remove the packing glands and use a different type of sealing mechanism, and 4) Initiated an evaluation of a procedure change to provide for backup manual closure of the check valves. In addition, the licensee initiated a computer calculation to determine the peak containment pressure resulting from a main steam line break inside containment, concurrent with a failure of one MSIV to close, and with less than 775 ft-lbs of steam flow force on the non-return check valve such that it would remain open. The licensee's calculation determined that containment pressure would peak at 55 lbs., which was less than the design pressure of 60 lbs. Based on that calculation, the licensee concluded that the main steam non-return check valves were currently operable and not in a nonconforming condition.

In response to the team concerns regarding not performing safety evaluations for test procedure changes, the licensee initiated AR 99-1000, "Potentially Inadequate 50.59s for Changes to PT-2.10.15." This AR addressed PCN 92T-0127, PCN 93-4130, and PCN 99-4171 and noted that they had not appropriately addressed the fact that a smaller than design basis steam line break could result in the blowdown of more that one steam generator. The AR also noted that a required 50.59 safety evaluation was not always included and that the UFSAR had not been updated.

In addition, the licensee also initiated AR 99-0959, "Action Report 99-0890 on Main Steam Line Check Valve Should Have Classified Condition as Nonconforming." This AR was to address potential weaknesses in the areas of Action Reporting, Operability Determination process, and related training deficiencies.

c. Conclusion

In general, the operability determinations reviewed were acceptable. A few of the operability determinations reached an appropriate conclusion, but were not thoroughly documented. One operability determination, regarding the main steam non-return check valves was inadequate.

The assumptions, analytical methods, and calculations used by the licensee to declare the main steam non-return check valves operable may not be conservative and may not be applicable in all cases. The licensee did not show that the uncertainty in the calculation is less than the available margin of torque needed to close the valve. Therefore, operability of the main steam non-return check valves remains an open issue pending NRC review of additional information from RG&E (See Attachment 2 of this report for additional questions).

The team identified several inadequate safety evaluations related to changes made to the main steam non-return check valves. Specifically, the valves were changed from free swinging gravity closing valves (as stated in the Updated Final Safety Analysis Report) to valves that required a substantial and increasing external force to close them, without addressing potential effects on steam generator integrity, containment integrity, steam generator tube integrity, reactor reactivity, or reactor vessel integrity. Other procedure changes failed to include safety evaluations. The team believes that changing the main steam non-return check valves to require a significant breakaway closing torque represents an Unreviewed Safety Question. This is an apparent violation of 10 CFR 50.59. **(EEI 50-244/99-05-01).** At the exit meeting on May 28, 1999, the licensee did not agree that these changes introduced an Unreviewed Safety Question.

E7.2 Onsite and Offsite Review Committees

a. Inspection Scope (40500)

The inspectors reviewed meeting minutes, attended onsite and offsite review committee meetings, interviewed committee members, and reviewed action tracking systems to determine the extent of committee involvement, oversight, and independence.

b. Observations and Findings

The team noted that members of both the onsite Plant Operations Review Committee (PORC) and the offsite Nuclear Safety Audit and Review Board (NSARB) asked good questions and initiated action items which were adequately tracked. During a PORC meeting, every member contributed substantially, indicating that they were both knowledgeable and prepared.

Questions Regarding Ginna's Main Steam Non-Return Check Valves

A review by the NRC of Calculation DA-ME-92-147, Rev. 2, dated 5/27/1999, "Main Steam Non-Return Check Valve Closure Analysis," for the Ginna Station of Rochester Gas and Electric Corporation, indicates that the assumptions, analytical methods, and calculations are not conservative and may not be applicable in all cases.

The NRC review observed that the steam flow past the check valve, with flow reversal occurring at the time of the incident, presents a very complicated flow geometry and the detail flow field and pressure distribution on the valve is needed to properly analyze the effects on the check valve. The licensee must show that the uncertainty in the calculation is less than the available margin of torque needed to close the valve. This needs to be demonstrated for breaks less than full double-ended guillotine breaks such that it represents the worst conditions regarding steam line break size and associated flow past the check valve attempting to close it. Lower flow rates from a less than full break would put even less closing torque on the valve.

Because of the NRC concerns, the following questions are provided:

1. What analysis method will be used? It is felt that a 3-dimensional computational fluid dynamics code is needed to properly model this complex flow condition. How will the analysis performed be validated for this type of application?
2. What is the basis for assuming that the steam flow is non-compressible?
3. Since flow in the line is changing in mass flow rate and reversing direction, what is the basis for assuming constant pressure (during normal operation the flow past the check valve is about 914 lb_m/sec.; then, subsequent to the line break the flow at the check valve reverses and decreases to 603.3 lb_m/sec.)?
4. Is the mass flow rate of 603.3 lb_m/sec in the calculation based on choked flow at the exit?
5. How was the mass flow rate coming from the 'line break' SG considered in the calculation of the 603.3 lb_m/sec coming from the 'operational' Steam Generator?
6. What is the basis for assuming the check valve closes in one (1) second?
7. What is the basis for the check valve disc being treated as a flat circular disc? Won't there be flow on both sides of the disc since the disc is round with gaps between the disc and the valve body?
8. What are the area and dimensions of clearance between the open disk circumference and the valve body? This information is needed to determine the area that is available for steam flow to exit the space above the open disk. And please provide, if readily available, in conjunction with your analysis,
   - the cross-sectional area:
   - for steam flow to enter the area above the open disk,
   - inside the inlet pipe to the valve,
   - at the most flow restrictive point inside the open check valve (e.g., the minimum throat area),
   - inside the outlet pipe from the valve,

   and, the volume:
   - above the disk,
   - in the valve body upstream of the minimum throat area,
   - in the valve body downstream of the minimum throat area.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

August 23, 1999

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

Reference: NRC #40500 Team Inspection 50.244/99-05 dated 8/6/99

Dear Mr. Vissing:

As a result of questions regarding main steam check valve performance included in the above reference, received August 16, 1999, RG&E and the NRC had a conference call on August 16 to review our approach in responding to these questions. A summary of the conference call is provided as Attachment 1.

As requested, we will formally respond to the questions within 30 days of receipt of that letter.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

U.S. NRC Ginna Senior Resident Inspector

Greg Cranston

ATTACHMENT 1: SUMMARY OF PHONE CONVERSATION W/NRC

DATE: 8-16-99
TIME: 10:00 AM

TOPIC: NRC 8 QUESTIONS ON Main Steam Check Valve

PARTICIPANTS: G. Wrobel (RG&E - Licensing)
J. Dunne (RG&E - Reactor Engineering & Analysis)
K. Muller (RG&E - Primary Systems)
G. Cranston (NRC - Region 1)

During the 8-16-99 phone conversation, RG&E reviewed its plan for responding to the 8 questions provided to RG&E by the NRC concerning the RG&E Design Analysis on Main Steam (MS) check valve closure on reverse flow following a MSLB. A synopsis of the information verbally discussed is summarized below.

Question 1

To respond to Question 1 on the need for a more sophisticated flow analysis, RG&E identified that it is planning to have a third party Independent Review of the issue. RG&E has had discussions with Duke Engineering & Services for performing the Independent Review. Duke has been provided with the RG&E Design Analysis (DA-ME-92-147, Rev.2) along with a copy of the valve drawing and the list of the 8 NRC questions. RG&E specifically has asked Duke to perform their review in terms of providing a response to both Question 1 and Question 7.

After obtaining the results of the Duke Independent Review and assessing their findings, RG&E will evaluate if any additional actions are believed to be warranted. This would include the need for a detailed 3D analysis to address the NRC concerns as well as the need to implement any changes to the present check valve configuration on a short term basis.

Question 2

The Design Analysis assumed that the change in steam density associated with the change in steam pressure was negligible for the pressure variations used in the analysis. Specifically, for an assumed steam pressure at the inlet of the check valve (e.g. 800 psia), the difference in density between static and stagnation conditions was negligible and could be ignored. This is based on the difference between static and stagnation conditions in the design analysis being on the order of a couple of psi. Attachment 1 of Design Analysis DA-ME-97-147 lists torques for various steam pressures of 700 psia, 800 psia and 900 psia. The incompressible assumption was not used to develop the pressure dependent torques listed in the Design Analysis; a curve fit was used for steam density as a function of the three assumed steam pressures.

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AREA CODE 716-546-2700

ROBERT C. MECREDY
Vice President
Nuclear Operations

September 24, 1999

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

Subject: Response to Questions Related to Main Steam Check Valve Performance per NRC Inspection 99-05

Reference: August 23, 1999 letter from Robert C. Mecredy to USNRC, "NRC #40500 Team Inspection 50-244/99-05, dated 8/6/99"

Dear Mr. Vissing:

On August 23, 1999, RG&E provided a summary of the discussions between RG&E and NRC personnel regarding main steam check valve performance questions arising from NRC Inspection 99-05 (see Reference). At that time we stated we would provide more detailed responses following the completion of an independent assessment being performed by Duke Engineering and Services. The attached responses include the results of that assessment.

We have concluded that the main steam check valves are operable, in that they would perform their safety function for the limiting steam line break, with substantial margin. We have also decided to initiate engineering activities to optimize packing of the valves so as to provide the minimum amount of friction needed for a leak-tight packing configuration. Recommendations from these engineering activities would be implemented during the year 2000 refueling outage.

Very truly yours,

Robert C. Mecredy

Attachment

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
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U.S. NRC Ginna Senior Resident Inspector

RESPONSE TO NRC QUESTIONS

QUESTION 1:

What analysis method will be used? It is felt that a 3-dimensional computational fluid dynamics code is needed to properly model the complex flow conditions. How will the analysis performed be validated for this type of application?

RESPONSE:

Based upon the introductory discussion to the eight NRC Questions, it appears that the major concern related to the RG&E Main Steam (MS) check valve analysis performed in Design Analysis DA-ME-92-147, Rev.2 (Reference 1) revolves around the fact that a fundamental change was made to the check valve without a comprehensive safety evaluation. This change resulted in a small difference between the calculated available torque due to reverse flow and required torque to initiate valve closure. Since the difference in available and maximum measured As-Found torque reported in Reference 1 was only approximately 1% (912 ft-lb vs 900 ft-lb), a concern exists with the uncertainty associated with the simplified methodology used in Reference 1 to quantify a complicated flow condition. Specifically, it has been stated that the licensee must demonstrate that the uncertainty in the Reference 1 calculation must be less than the available margin of torque needed to close the valve.

RG&E concurs with the NRC assessment that the flow pattern around the MS check valve disk under reverse flow conditions represents a complicated flow geometry; however, RG&E believes that sufficient conservatisms exist in the Reference 1 approach to bound these uncertainties. Specifically, the following areas of conservatism exist with the RG&E methodology that causes its calculated torque value to be significantly below the true torque that would be generated during a reverse flow condition for the limiting MS break size:

- Conservative reverse mass flow rate
- Conservative static pressure under the valve disk
- Conservative break size selection

The conservatism resulting from these three areas are discussed below.

Mass Flow Rate

The major source of conservatism in the Reference 1 analysis involves the reverse mass flow rate that was assumed. The mass flow rate used in Reference 1 to assess closing torque was obtained after reviewing the LOFTRAN analysis performed in Reference 2. In lieu of using transient reverse flow and MS pressure data from the LOFTRAN analysis, the check valve analysis used a single mass flow rate and MS pressure that bounded the LOFTRAN transient data. If the transient reverse flow data had been used to generate a time dependent torque curve, the Reference 1 methodology would have calculated significantly higher transient torque values than what was calculated in Reference 1.

To demonstrate this condition, a revised LOFTRAN analysis was performed in Reference 3 that more closely modeled the reverse flow transient for mass flow from the intact SG to the break location. The Reference 3 analysis also used a more conservative assessment of MS piping hydraulic resistance to minimize reverse flow from the intact SG. The resulting transient SG pressures for the limiting 0.86 ft^2 steam line break is shown in Figure 1. The resulting transient mass flow rate contribution to the total break flow from each SG is shown on Figure 2.

The Figure 2 results indicate that no reverse flow exists through the MS check valve until after the turbine stop valves have closed. Prior to the closure of the turbine stop valves, all of the break flow is supplied from the faulted SG. After the turbine stop valves have closed, the break area represents the only flow path available for both the faulted and intact SGs. Consequently, reverse flow is initiated from the intact SG to the break location immediately following the isolation of flow to the turbine. The transient flow distribution to the break from the two SGs is a function of the individual SG pressures and the hydraulic resistances for the two flow paths.

Due to the more rapid de-pressurization of the faulted SG prior to the turbine stop valve closure, the intact SG pressure is higher than the faulted SG. This pressure difference causes a surge of flow from the intact SG to the break immediately following the stop valve closure as the difference in SG pressures adjust to the new flow network represented by the closed stop valves and the break. If the initial surge did not close the check valve, the differences in SG pressure would decrease and the flow from the faulted SG would increases while the flow from the intact SG decreases until a new quasi-equilibrium condition exists as shown in Figures 1 and 2.

The maximum reverse check valve flow shown in Figure 2 is approximately 46 % higher than the flow rate used in the Reference 1 analysis. This higher flow indicates that appreciable margin exists with the Reference 1 calculated torque. Using the transient reverse flow and SG pressure data from Figures 1 and 2, the transient torque developed across the check valve disk with the Reference 1 methodology has been calculated in Reference 4. These results are shown in Figure 3. The Figure 3 results indicate that immediately following the turbine stop valve closure, the initiation of reverse flow from the intact SG to the break results in a calculated torque value that would exceed 2000 ft-lb. This initial torque is more than a factor of two higher than the value calculated in Reference 1. Therefore, the Figure 3 results demonstrate that the mass flow rate and MS pressure used in the Reference 1 calculation were chosen in a conservative manner.

Valve Static Pressure

A second major conservatism in the Reference 1 methodology is the static pressure assumed under the valve disk. The major contributor in the Reference 1 analysis to the valve closing torque is the differential pressure assumed across the valve disk. The differential pressure term used in Reference 1 was approximately 1.3 psi; and, this pressure difference generated approximately 95 % of the total torque calculated by Reference 1.

The differential pressure across the disk is the difference between the static pressure on the top side of the disk and the static pressure on the bottom side of the disk. For the static pressure on the top side of the disk, Reference 1 assumed the top side of the disk would represent a low flow region. Therefore, the static pressure on the top side of the disk was assumed to approach the fluid stagnation pressure. Since any flow through the top side of the disk results in a static pressure that is less than the fluid stagnation pressure, this assumption is inherently non-conservative. However, the difference between static and stagnation pressure on the top side of the disk was judged to be small and was more than compensated by the conservative assessment of static pressure under the disk made by Reference 1.

Reference 1 assumed that the fluid static pressure under the valve disk was equal to the fluid static pressure in the piping upstream of the MS check valve. The fluid static pressure in the upstream piping was calculated based on the piping cross sectional area of 593 in^2 based on the 27.5" pipe ID. As the steam flows into the MS valve body and flows underneath the valve disk, the valve flow area decreases; thereby causing the steam velocity to increase and its static pressure to decrease. Due to the orientation of the valve disk when it is up against its stop and due to the valve design (as shown on Reference 5), the flow area continues to decrease as it travels from the leading edge of the disk to the point just upstream of the valve seat area. As the steam enters the valve seat area and clears the back end of the disk, the steam flow area increases. The flow area at the valve seat location is approximately 452 in^2 based upon the seat ID of 24" specified by Reference 5.

Although the 24" seat ID is not the minimum under disk flow area, it can be used to estimate the magnitude of the change in static pressure from the valve inlet through to the seat area location. By conservatively ignoring frictional losses associated with the check valve flow, the decrease in static pressure between the valve inlet and the valve seat area can be assumed to equal the increase in the velocity head between these two locations. The velocity head in turn is proportional to the square of the flow velocity(or inversely proportional to the square of the flow area). Consequently, for a 23.8 % decrease in flow area between the valve inlet and the valve seat location, the velocity head term would increase by 53.2 %. At the valve inlet, the velocity head term as calculated in Reference 1 was approximately 1.3 psi. Therefore, at the valve seat area, the velocity head term would be approximately 2 psi. This would result in a decrease in the static pressure under the disk of approximately 0.7 psi(2.0 psi - 1.3 psi).

Since the majority of the check valve flow will occur under the valve disk, the 0.7 psi magnitude decrease in static pressure on the underside of the valve disk is more than sufficient to compensate for any non-conservatisms introduced into the Reference 1 analysis due to the stagnation assumption for the valve area above the disk. This magnitude change in static pressure for the underside disk area would ensure that for the flow conditions analyzed in Reference 1 that the actual static pressure differential across the valve disk would be greater than the approximately 1.3 psi value used to calculate closing torque.

Break Area

A third area of conservatism in the Reference 1 analysis relates to the break size assumed for the limiting MSLB where operation of the check valve under reverse flow conditions is assumed to occur. Reference 2 evaluated both the containment pressure response and the RCS response to a 0.86 ft^2 main steam line break (MSLB). For this break size with no closure of the MS check valve both the peak containment pressure and the RCS core response were within the design basis conditions for Ginna Station. The peak containment pressure calculated for this MSLB was approximately 59 psig. Since this is below the containment design pressure of

60 psig, this break size was chosen as the threshold break size for evaluating valve closure torque in Reference 1.

Since the peak calculated containment pressure for the 0.86 ft^2 break is below the containment design pressure of 60 psig, it represents a conservative choice for the threshold break size. If additional iterations on peak calculated pressure as a function of break size had been performed, it would have been possible to justify a somewhat larger break size that would have still kept containment pressure below its 60 psig design value. The larger break size would have resulted in higher break flow rates; and, correspondingly, higher check valve reverse flow rates and disk torques. Although the increase in total reverse flow that would have occurred is expected to be small, it does represent an additional conservatism in the choice of mass flow rate used by RG&E in Reference 1 to analyze valve closing torque as a result of reverse flow.

<u>RG&E Alternate Calculation</u>

To perform a check on the adequacy of the Reference 1 methodology for determining valve torque, RG&E in Reference 4 also evaluated the valve closure torque that would result solely as a function of frictional differential pressure across the valve disk. Since most of the frictional pressure drop is expected to be due to losses associated with the valve disk, the overall check valve hydraulic resistance can be used as a means for checking the adequacy of the Reference 1 methodology.

From the original Bill of Material for the MS check valves the design differential pressure at 100 % power conditions with forward flow is 2.72 psi. Using this differential pressure and the 100 % power MS conditions for flow rate and pressure, Reference 4 calculated the hydraulic resistance for the valve for forward flow. For reverse flow condition, the check valve hydraulic resistance would be larger than that observed under forward flow conditions. The increase in hydraulic resistance for the valve would result primarily from the leading edge effect associated with the valve disk sitting on its stop. Since the leading edge of the valve disk under reverse flow protrudes approximately 2" into the flow stream, the disk would create increased turbulence and corresponding frictional losses under reverse flow.

Reference 4 conservatively used the forward flow hydraulic resistance for calculating frictional differential pressure across the valve disk under reverse flow conditions. The resulting differential pressure as a function of time based upon the Figure 1 and Figure 2 SG pressure and check valve flow rates was then calculated. This differential pressure was then used to calculated the net load on the valve disk and the corresponding closing torque. The results of this calculation are shown on Figure 4, where it is compared to the transient Reference 1 methodology results previously discussed.

The alternate methodology based on frictional pressure drop shows a transient profile that is similar to the Reference 4 transient methodology results. The calculated torque values are approximately 18 % lower than the Reference 4 transient methodology; however, its results are still significantly higher than the torque value used in the Reference 1 static analysis. The difference with the Reference 4 transient results is attributed primarily to the following two conservatisms associated with the alternate methodology:

1. Use of forward flow hydraulic resistance for reverse flow.
2. Neglecting difference in static pressure differences between the top and bottom side of the valve disk

Therefore, although the closing torques calculated by the alternate methodologies are lower than those obtained with the Reference 1 methodology; they also demonstrated that at the beginning of reverse flow conditions the closing torque on the valve disk is appreciably higher than the 900 ft-lb required to initiate valve closure.

On-Going Activities

In addition to the information provided above RG&E has a number of on-going activities related to this issue. These activities include:

1. Independent Third Party Review of Valve Torque
2. Assessment of Means to Reduce Closure Torque

Third Party Review

As a result of the concern identified with the adequacy of the RG&E Reference 1 method for determining valve closure torque, RG&E has requested that Duke Engineering & Services (DE&S) perform an independent third party review of this issue. The results of the DE&S Independent Review are documented in Reference 6 and are summarized below.

RESPONSE TO NRC QUESTIONS

Although DE&S performed a literature search for experimental data on swing check valve closing torque; no relevant information was found. Consequently, DE&S analytically assessed check valve closure torque based upon two alternate methodologies. One method used information for assessing torque on closure of tilting disk check valves; whereas, the second methodology used information for closure of butterfly valve disks. Valve closure torques were calculated for reverse flow rates that varied from the 603 lb/sec value used in Reference 1 to the maximum flow rate shown on Figure 2. For both methods conservative valve characteristics were chosen. When compared to the original RG&E method used in Reference 1, the two methods calculated torques that were respectively 10 % and 33 % less than the RG&E method.

Although the DE&S alternate methods calculated lower torques at 603 lb/sec than was used by RG&E in Reference 1; DE&S identified that the torque developed by the actual transient flow shown in Figure 2 resulted in maximum torques well in excess of the 912 ft-lb value calculated by Reference 1. Actual torque margins based upon transient flows ranged from 43 % to 91 % for the two alternate methods. Additionally, DE&S stated that the rapid increase in reverse flow experienced by the check valve would result in a transient impact loading on the valve packing that would cause valve movement at a lower torque than would be developed during normal valve torque testing. Based upon the large flow margin available between the flow used by RG&E in Reference 1 and the actual transient flow shown in Figure 2, DE&S concluded that the fluidynamic forces experienced by the check valve would be sufficient to close the check valves when experiencing the transient flow rates shown on Figure 2.

Finally, although the fluid flow may be sufficient to cause valve closure; DE&S stated that the present 600 ft-lb torque value used by RG&E to establishing packing compression is excessive based upon their experience with swing check valves. Consequently, DE&S recommended that the valve and packing configuration be reviewed and reworked as necessary so as to lower the packing torque used to set up the valve.

Reduction of Closure Torque

As a result of the on-going discussion and questions related to this issue between RG&E and the NRC, RG&E believes that it is prudent to reduce the torque required to initiate valve closure in order to return the valve to a condition more representative of the original design intent (i.e. gravity closing). In order to accomplish this, RG&E has relocated the check valve counterweights to their fully retracted position. This decreased the moment arm associated with the counterweights by approximately six inches.

Since the two 150 lb counterweights act to prevent valve closure, their relocation has decreased the amount of torque required by reverse flow to initiate valve closure by approximately 150 ft-lb. For the nominal 600 ft-lb set-up torque used for establishing valve packing friction coming out of the 1999 Refueling Outage, this results in a 25 % reduction of the required flow induced torque to 450 ft-lb. For the largest As-Found measured torque of 900 ft-lb, the 150 ft-lb reduction decreases the required torque due to reverse flow by approximately 17 % to 750 ft-lb.

In addition to this short term action, RG&E is reviewing other long term actions that would be implemented in the 2000 Refueling Outage to decrease the torque required to initiate valve closure under reverse flow conditions. These actions include the complete removal of the counterweights as well as changes in the method used to pack the valve.

Conclusion

Based upon the conservatisms discussed above and the results of the independent assessment performed of valve closure torque, RG&E concludes that sufficient margin exists between calculated torque and the maximum As-Found measured torque to ensure that closure of the Main Steam check valves would occur under reverse flow for the most limiting Main Steam Line Break. The most conservative analytical assessment discussed above provides greater than 40 % margin to the maximum As-Found measured torque of 900 ft-lb. Due to this large amount of margin, RG&E concludes that a three dimensional computational fluid dynamics analysis of the check valve is not warranted.

To provide additional margin for present and future plant operation, RG&E has initiated actions to reduce the actual breakaway torque that would be needed for check valve closure. For present plant operation RG&E has re-positioned the valve counterweights to minimize the torque that acts to prevent valve closure. This action has decreased the torque needed to intitate valve closure by approximately 150 ft-lb. For the present operating cycle this activitity has provided, as a minimum, an addition 17 % of torque margin. Finally, for future operating cycle RG&E has initiated engineering activities to optimize packing of the valve so as to provide the minimum amount of packing friction needed for a leak tight packing configuration. Recommendations from these engineering activities would be implemented during the year 2000 Refueling outage.

FIGURE 1 - SG PRESSURE

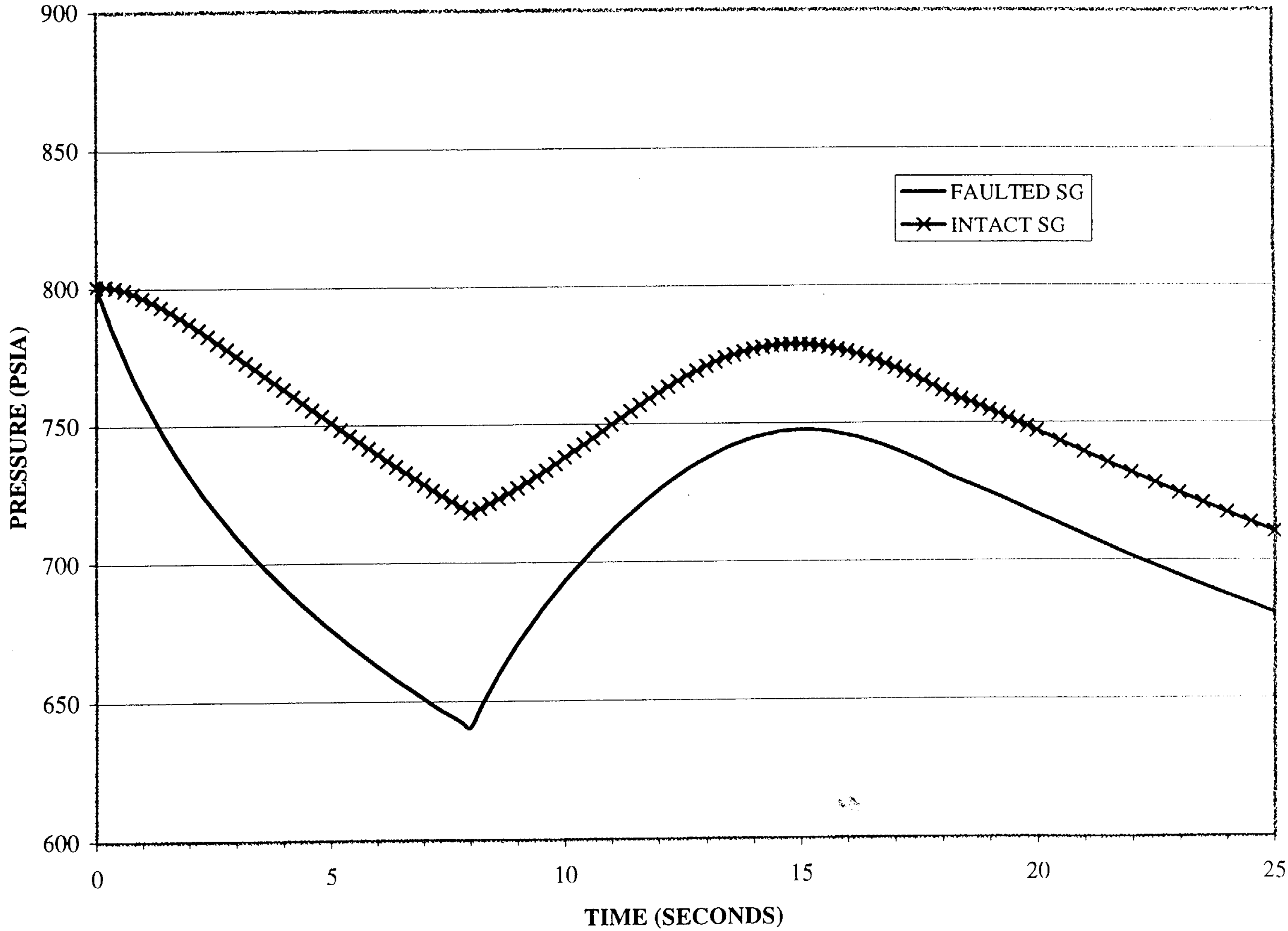


FIGURE 2 - BREAK FLOW DISTRIBUTION

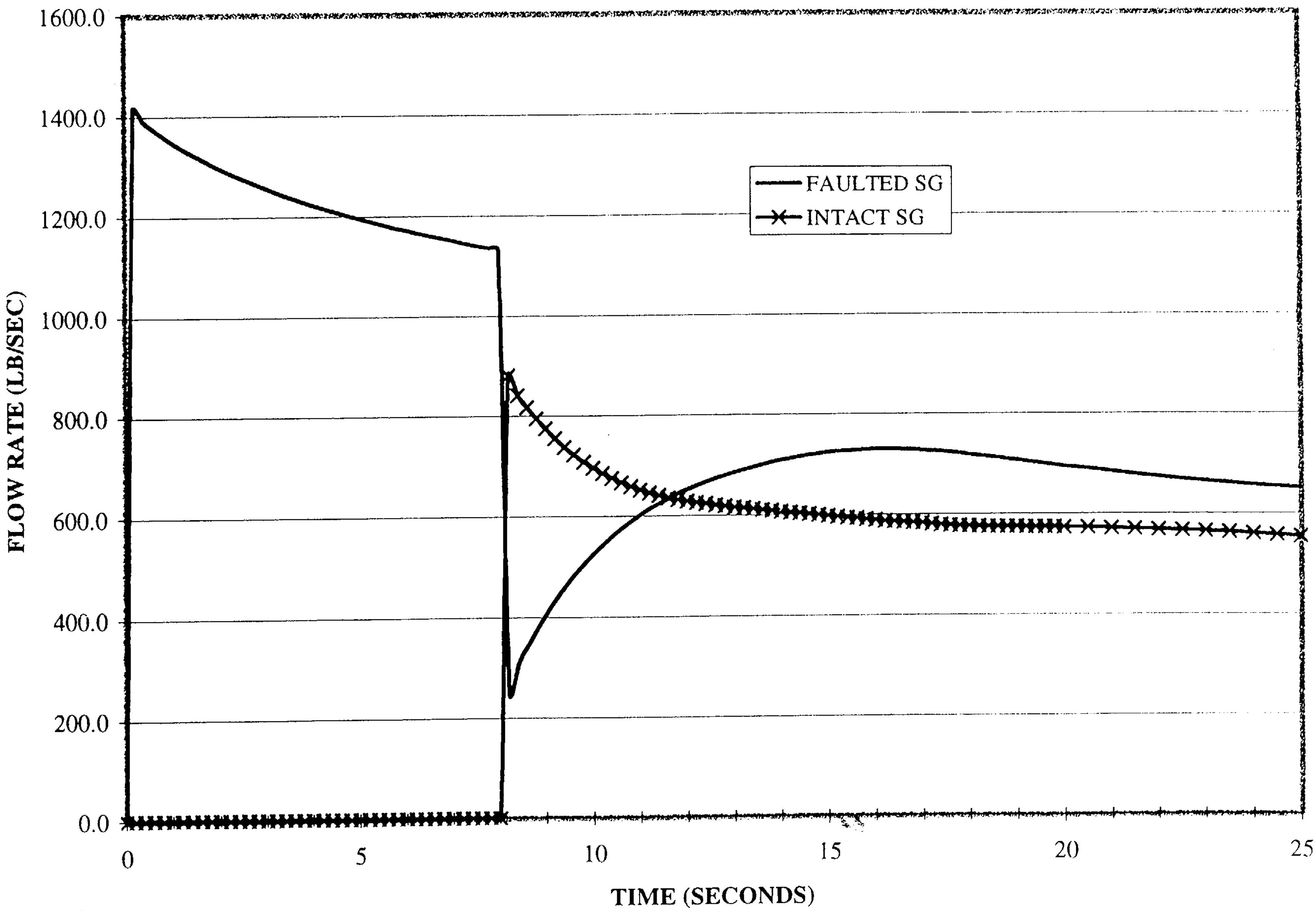
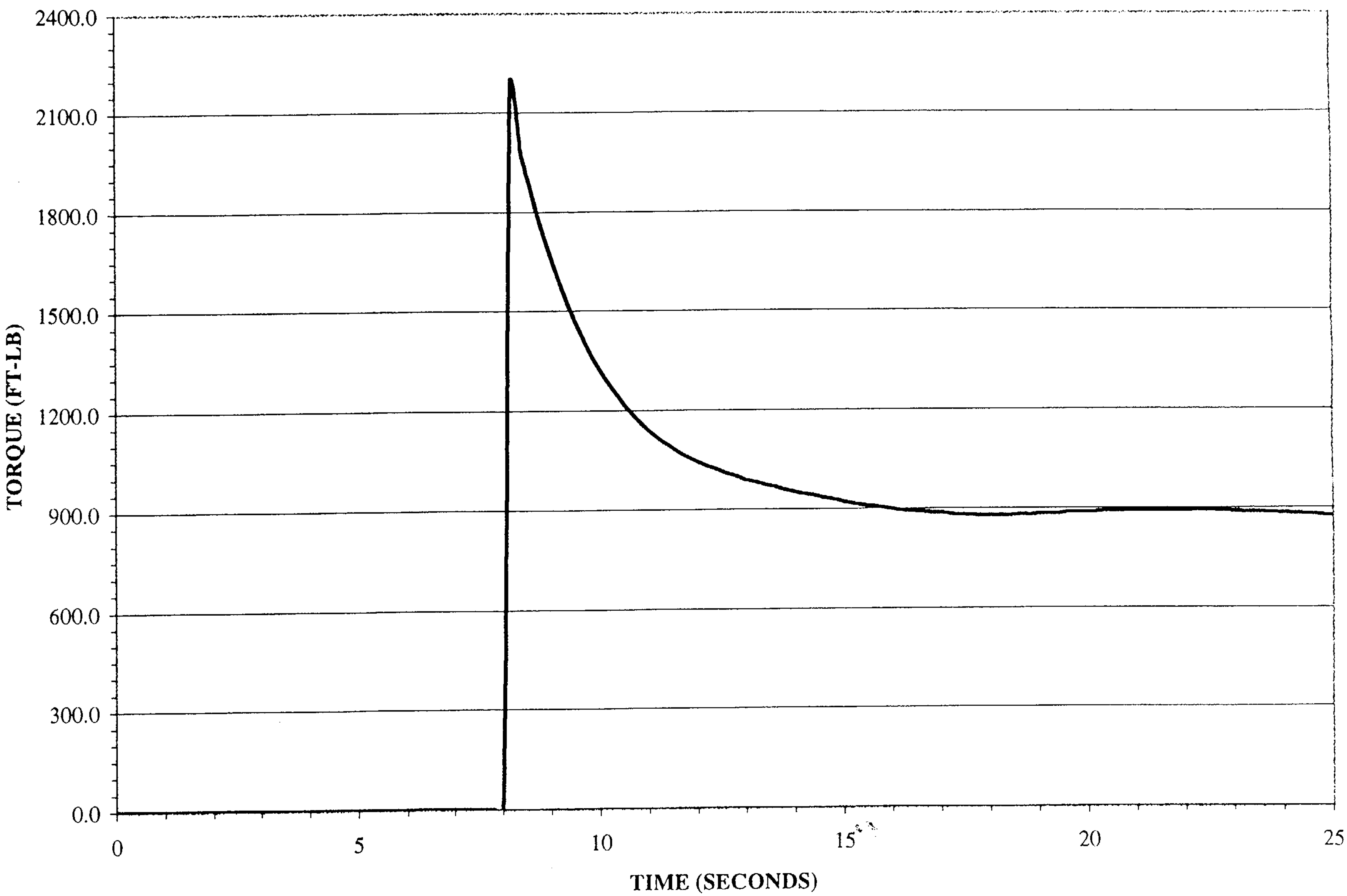


FIGURE 3 - CHECK VALVE TORQUE



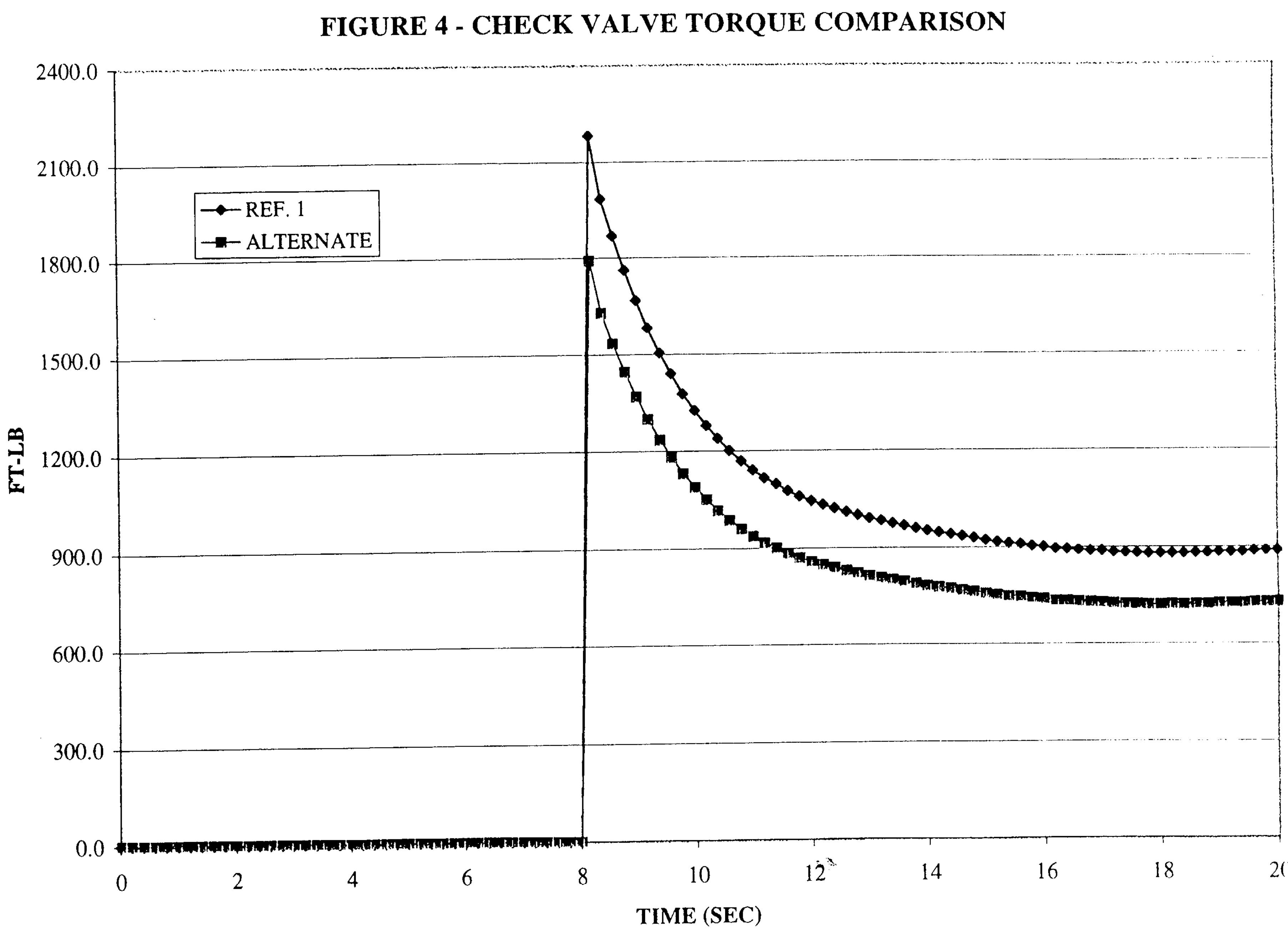


FIGURE 4 - CHECK VALVE TORQUE COMPARISON
REF. 1
ALTERNATE
FT-LB
2400.0
2100.0
1800.0
1500.0
1200.0
900.0
600.0
300.0
0.0
0
2
4
6
8
10
12
14
16
18
20
TIME (SEC)

RESPONSE TO NRC QUESTIONS

QUESTION 2:

What is the basis for assuming that the steam flow is non compressible?

RESPONSE:

In Section 5.1 of RG&E Design Analysis DA-ME-92-147, Revision 2; it is stated that the saturated steam flow through the check valve is assumed to be non-compressible since the pressure is relatively constant across the valve. The Design Analysis evaluates the closing torque generated at a specific point in time where the reverse flow rate through the check valve is 603.3 lbm/sec and the steam pressure at the valve is approximately 800 psia. The Design Analysis then calculates the pressure head associated with the steam velocity at the inlet to the check valve. Since the steam velocity pressure head term is small (e.g. approximately 1.3 psi), the difference between the fluid static pressure and stagnation pressure is approximately 0.15 % (1.3 psi / 800 psia). For steam velocities with Mach numbers less than 0.1 (e.g. velocity less than approximately 150 ft/sec), isentropic flow relationships for an ideal gas indicates that the difference between static and stagnation densities are less than 0.5 %. Therefore, the density change for a compressible fluid for the fluid conditions that would exist in the valve body are small and can be neglected. This represents the basis for the non-compressible assumption made in Section 5.1 of the Design Analysis.

It should be noted that Attachment 1 of Design Analysis DA-ME-92-147, Revision 2 includes a plot of torque versus steam flow for main steam line pressures of 700 psia, 800 psia and 900 psia respectively. The non-compressible assumption was not used to develop the torque valves calculated for these three steam pressures. For each steam pressure (e.g. 700 psia, 800 psia and 900 pisa) the corresponding saturated steam density was used to determine torque as a function of steam flow rate. The non-compressible assumption was only used for the calculation of the velocity pressure head for each flow condition and steam pressure.

RESPONSE TO NRC QUESTIONS

QUESTION 3:

Since flow in the line is changing in mass flow rate and reversing direction, what is the basis for assuming constant pressure (during normal operation the flow past the check valve is about 914 lbm/sec; then, subsequent to the line break the flow at the check valve reverses and decreases to 603.3 lbm/sec)?

RESPONSE:

Design Analysis DA-ME-92-147, Rev.2 used a check valve flow and steam pressure at one point in time to calculate the corresponding closure torque developed by the flow and pressure conditions. This pressure and flow were chosen to bound the transient data. If the transient flow and pressure data were used, a transient torque curve could have been generated that would take into account the time dependent nature of the flow and pressure experienced by the check valve. This transient torque data has been provided in response to Question 1 and, it demonstrates the inherent conservatism in choosing a single bounding point.

Additionally, once sufficient torque is developed to overcome the valve packing friction, the resulting movement of the check valve disk into the flow stream would result in an increase in drag force across the valve disk which would ensure that the valve would go closed. Therefore, DA-ME-92-147, Rev. 2 did not need to evaluated flow and pressure conditions that would exist subsequent to the initiation of valve closure.

RESPONSE TO NRC QUESTIONS

QUESTION 4:

Is the mass flow rate of 603.3 lbm/sec in the calculation based on choked flow at the exit?

RESPONSE:

The LOFTRAN computer program was used in DA-NS-99-054, Rev. 0 (Reference 2)to calculate the blowdown of the Steam Generators due to a steam line break. The LOFTRAN program calculated the transient flow at the break location as well as the transient flow supplied to the break by both Steam Generators. The break flow rate represents the summation of the two flow paths that feed the break. The actual total break flow is determined by use of a choked flow correlation for saturated steam. The choked break flow is primarily a function of both the break area size and the main steam line pressure at the break location.

RESPONSE TO NRC QUESTIONS

QUESTION 5:

How was the mass flow rate coming from the "line break" SG considered in the calculation of the 603.3 lbm/sec coming from the "operational" Steam Generator?

RESPONSE:

The flow rate out of the break at any point in time is determined based upon choked flow, the break size and the local steam line pressure at the break location. The break flow is fed by flow that reaches the break from both SGs after the turbine stop valves have closed. Consequently, the break flow represents the summation of the two individual flow paths. The transient flow rates for the two flow paths that supply the break are shown in Figure 2.

Prior to closure of the turbine stop valves, the flow rate exiting the faulted SG exceeds the break flow (i.e. a portion of the steam flows to the turbine). Therefore, all of the flow out of the break is supplied by the faulted SG up to the time that the turbine stop valves close. No flow from the intact SG reaches the break until after the turbine stop valves close. The 603.3 lb/sec value used in the determination of valve closure torque under reverse flow conditions was chosen to be a conservative assessment of the reverse flow conditions that would exist for the check valve.

RESPONSE TO NRC QUESTIONS

QUESTION 6:

What is the basis for assuming the check valve closes in one (1) second?

RESPONSE:

Section 7.1 of Design Analysis DA-ME-92-147, Rev. 2 states that the mass flow and pressure conditions at t = 1 second were used since this is the check valve closure time assumed. The one second time represents the typical UFSAR Chapter 15 accident analysis time for Main Steam check valve closure following initiation of reverse flow from a design basis double ended guillotine rupture. Consequently, the mass flow rate of 603.3 lbm/sec and 800 psia represent the LOFTRAN calculated values for flow from the "intact" SG at the one second time in the main steam line break transient as calculated by DA-NS-99-054, Rev. 0. As stated in Section 5.3 of DA-NS-99-054, Rev. 0; the use of the flow and pressure at 1 second into the transient is conservative since the actual flows and pressures that would exist following the Turbine Stop valve closure generate higher valve closure torques. This has been demonstrated by the transient torque curve provided in response to Question 1.

In reality for this smaller steam line break, reverse flow through the check valve from the "intact" SG would not occur until after the Turbine Stop Valves have closed terminating flow from the two SGs to the Turbine. This would occur after 1 second in time. This has been demonstrated by the transient flow and pressure results provided for the response to Question 1 (Figures 1 & 2).

The actual value of one second has no significant impact on the DA-ME-92-147, Revision 2 analysis.

RESPONSE TO NRC QUESTIONS

QUESTION 7:

What is the basis for the check valve disc being treated as a flat circular disc? Won't there be flow on both sides of the disc since the disc is round with gaps between the disc and the valve body?

RESPONSE:

The assumption made in Reference 1 of treating the check valve as a flat disk was used to determine an appropriate drag co-efficient for steam flow over the valve disk. The drag co-efficient was then used to calculate an appropriate drag force and corresponding moment. The flat disk was used since the front edge of the valve disk that sits in the flow stream under reverse flow is a circular disk with a thickness of approximately 3.75" as shown on Reference 5. The bottom side of the disk is flat over its entire surface. The top side of the disk is flat over approximately the first half of the disk.

At the top side center of the disk the disk hinge arm is attached to the disk by a hex nut. Any flow above the disk over the back half of the disk will also experience interaction due to the presence of the disk arm. Since the disk arm and the arm hex nut connection at the center of the disk provide a flow obstruction for flow on the top side of the disk, their presence would contribute to increased drag on the disk. Therefore, it was judged that ignoring their presence and treating the disk as a flat circular disk was conservative for assessing an appropriate drag co-efficient for the valve disk. Additionally, with the valve disk full open up against it stop, the valve disk presents a 15° negative angle of attack (angle to flow stream below horizontal orientation) under reverse flow conditions. Due to this negative angle of attack, the valve disk would generate a drag load that would act on the valve body in a direction that would cause it to go closed.

Since the leading edge of the flow disk protrudes approximately 2" below the top of the valve body ID, the portion of the valve flow above the valve disk would be expected to be scooped into the valve body area above the valve disk. The flow area above the disk is large in relationship to the valve flow area that would push flow above the valve disk. This large flow area would cause the steam velocity above the disk to be significantly below its value in the valve body. This reduction in velocity would cause the static pressure of the fluid above the valve disk to approach the fluid stagnation pressure.

RESPONSE TO NRC QUESTIONS

It should also be noted that Reference 1 conservatively neglected the projected area for the back half of the valve disk when assessing the drag force acting on the valve disk body. This decreased the total drag force calculated in Reference 1 by 50 %. Additionally, with the Reference 1 methodology the moment calculated for the drag force represents only a small percentage of the total calculated moment. Only approximately 5 % of the total moment calculated by Reference 1 results from the drag force calculation. Consequently, the impact on the flat disk drag co-efficient associated with the presence of the disk arm and hex nut attachment is expected to have a minimal impact on the torque calculation performed by Reference 1. Therefore, any uncertainty associated with the flat disk assumption is expected to be negligible; and, would be bounded by the conservatisms discussed in the response to Question 1.

RESPONSE TO NRC QUESTIONS

QUESTION 8:

What are the area and dimensions of clearance between the open disc circumference and the valve body? This information is needed to determine the area that is available for steam flow to exit the space above the open disc. And please provide, if readily available in conjunction with your analysis, the cross-sectional area:

- for steam flow to enter the area above the open disk,
- inside the inlet pipe to the valve,
- at the most flow restrictive point inside the open check valve (e.g., the minimum throat area),
- inside the outlet pipe from the valve,

and, the volume:

- above the disk,
- in the valve body upstream of the minimum throat area,
- in the valve body downstream of the minimum throat area.

RESPONSE:

RG&E presently has no quantitative information from either the vendor (Atwood-Morrill) or past on-site examinations on the clearances between the valve disk and the valve body. Based upon the vendor drawing (Reference 5) and the full open orientation of the valve disk, it is expected that the clearance varies along its entire circumference. The maximum gap dimension is expected to occur at the leading edge of the valve disk. The minimum clearance would be expected to occur at the hinge pin location.

With regard to the specific areas and volumes requested by the NRC, no quantitative information on volumes is presently available from the vendor drawing (Reference 5); however, since Reference 5 is a scaled drawing it may be possible to approximate the requested volumes be using scaled dimensions from the drawing. The quantitative cross sectional area information requested by the NRC based upon Reference 5 is listed below:

1.	Steam flow to enter the area above the open disk	Not Specified
2.	Inside the inlet pipe to the valve	594 in^2
3.	The most flow restrictive point inside the open check valve	Not Specified
3A.	Flow area at the valve seat location	452 in^2
4.	Inside the outlet pipe from the valve	594 in^2

RESPONSE TO NRC QUESTIONS

The results for items 2 and 4 are based upon the nominal piping inside diameter of 27.5" for the 30" Main Steam piping attached to the valve body. The weld prep details on Reference 5 support this dimension. The valve areas for flow to enter the valve top and for the minimum restriction location under the valve disk are not specified; however, the flow area for the valve seat location has been provided based upon the seat ID listed on Reference 5.

RG&E has discussed with Atwood-Morrill the availability of the information on valve disk clearances, valve areas and valve volumes as requested by the NRC. Presently Atwood-Morrill has stated that this information is unavailable. RG&E is pursuing with Atwood-Morrill the possibility of obtaining this information; however, its future availability is uncertain at this time.

RESPONSE TO NRC QUESTIONS

REFERENCES:

1. RG&E Design Analysis DA-ME-92-147, Rev. 2
2. RG&E Design Analysis DA-NS-99-054, Rev. 0
3. RG&E Design Analysis DA-NS-99-054, Rev. 1
4. RG&E Design Analysis DA-ME-99-070, Rev. 0
5. Atwood-Morrill Dwg 20729-H, Rev. 3B
6. Duke Engineering & Services Report RG0007-T14-001, "Assessment of Main Steam Non-Return Check Valve Closure Analysis", Rev. 0