Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150
Tel 601 437 2800

GNRO-2008/00048

June 05, 2008

U.S. Nuclear Regulatory Commission

Attn: Document Control Desk
Washington, DC 20555-0001

Subject: LER 2008-003-00

Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

Dear Sir or Madam:

Attached is Licensee Event Report (LER) 2008-003-00 which is a final report.

This letter does not contain any commitments.

Yours truly,

Michael J. Larson
Acting Licensing Manager

MJL:mjl
attachment: LER 2008-003-00
cc: (See Next Page)

cc: NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

U. S. Nuclear Regulatory Commission
ATTN: Mr. Elmo E. Collins (w/a)
Regional Administrator, Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

U. S. Nuclear Regulatory Commission
ATTN: Mr. Jack Donohew, NRR/DORL (w/2)
ATTN: ADDRESSEE ONLY
ATTN: U. S. Postal Delivery Address Only
Mail Stop OWFN/O-8G14
Washington, DC 20555-0001

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Grand Gulf Nuclear Station, Unit 1	05000 416	1 OF 8

4. TITLE Increased Buss Voltages Results in Breaker Trip on Over Current of a High Pressure Core Spray Pump Low Flow Valve Resulting in Non-Compliance with Technical Specification 3.6.1.3, Primary Containment Isolation Valve Function

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME: N/A	DOCKET NUMBER: N/A
04	10	2008	2008	- 003 -	00	06	05	2008	FACILITY NAME: N/A	DOCKET NUMBER: N/A

9. OPERATING MODE: 1

10. POWER LEVEL: 099

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Grand Gulf Nuclear Station - Michael J. Larson, Acting Licensing Manager	601-437-6685

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	BG	ISV	L200	Y					

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

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

On March 05, 2008 at 1619, Technical Specification Emergency Core Spray System (ECCS) surveillance testing was being performed of the High Pressure Core Spray (HPCS) pump and system. During the testing, the HPCS low flow valve 1E22-F012 (also a primary containment isolation valve), while stroking from closed to open position, de-energized and the HPCS loss or overload status light energized. The valve was found in the non-closed position and the power supply breaker for the motor actuator for the valve was found tripped open. This condition is considered a violation of Technical Specification 3.6.1.3 Primary Containment Isolation Valve (PCIV) due to exceeding the LCO Required Action Completion Time of four hours to isolate the penetration.

The cause of valve 1E22-F012 to fail to close is due to its supply breaker instantaneous over current trip settings being set too low, thus rendering the valve inoperable. Investigation revealed that this condition had existed since the early 1990s when buss voltages had been increased to a higher value to account for under voltage events.

Corrective actions were implemented which included replacement of the valve 1E22-F012 power supply breaker and increasing the instantaneous over current trip settings. Breaker settings for the other ECCS motor operator valves that were susceptible to this condition were checked and found to be acceptable.

A. REPORTABLE OCCURRENCE

On March 05, 2008 at 1619, with the plant in Mode 1 (Power Operation) at about 99 percent reactor power, Technical Specification (TS) Emergency Core Spray System (ECCS) surveillance testing was being performed of the High Pressure Core Spray (HPCS) [BG] pump and system. During the testing, the HPCS low flow valve 1E22-F012 (also a primary containment isolation valve) [ISV], while stroking from closed to open position, de-energized and the HPCS loss or overload status light energized. The valve was found to remain in non-closed position and the power supply breaker [52] for the motor actuator for the valve was found tripped open.

It was discovered that when the offsite power circuit voltage is >/= 521 kV, that the power supply breaker for valve 1E22-F012 could trip on over current, thus rendering the valve inoperable. This then could result in valve 1E22-F012 to fail to be closed from the control room. Investigation revealed that this condition had existed since the early 1990s, when buss voltages had been increased to a higher value to account for under voltage events. This condition is considered a violation of TS 3.6.1.3 Primary Containment Isolation Valve (PCIV) due to exceeding the Limiting Condition for Operation (LCO) Required Action Completion Time of four hours to isolate the penetration. A review of offsite voltages for the past three years indicates there were at least nineteen periods when the offsite power circuit voltage was >/= 521 kV for at least 1 day, therefore a condition prohibited by Technical Specifications has existed prior to discovery of the condition. This condition is reportable in accordance with 10CFR50.73(a)(2)(i)(B) and since TS 3.6.1.3 LCO Completion Time of four hours to close the valve was not met. The reportability event discovery date was April 10, 2008 at 0957 due to the time needed to properly evaluate the condition and affected Technical Specifications.

B. INITIAL CONDITIONS

At the time of the event, the reactor was in OPERATIONAL MODE 1 with reactor power at approximately 99 percent. There were no additional inoperable structures, systems, or components at the start of the event that contributed to the event.

C. DESCRIPTION OF OCCURRENCE

This condition was discovered while the HPCS pump was declared inoperable for required surveillance testing on March 05, 2008. During the surveillance testing, the HPCS low flow valve 1E22-F012 (also a primary containment isolation valve), while stroking from closed to open position, de-energized which was caused by a power supply breaker (52-170109) trip (HPCS MOV Overload/Power Loss status light energized in the Control Room). Valve 1E22-F012 was found in the non-closed position and its motor actuator supply breaker had tripped open on over current. It was discovered that when the offsite power circuit was >/= 521 kV, the valve could fail to close due to inadvertent supply breaker tripping on over current.

For reportability purposes the valve was considered inoperable during periods of time when the offsite voltage was >/= 521 kV which made the valve potentially capable of failing to close. To determine reportability, three years of offsite power circuit voltage data were reviewed between 2005 through 2008. Using this data, the period for reportability chosen was a Completion Time of 1 day, which would have exceeded TS LCO 3.6.1.3 Required Action A.1 Completion Time of four hours to close valve 1E22-F012. The data indicates that there were at least nineteen separate periods that the offsite power circuit was >/= 521 kV. For Technical Specification LCO 3.6.1.3 a period of 1 day with the offsite power circuit voltage >/= 521 kV would have made the 1E22-F012 PCIV inoperable and thus exceed its required TS Completion Time of four hours to isolate the valve. Since the valve could fail to close, it was considered inoperable, thus compliance with TS LCO 3.6.1.3 would have been warranted for periods that the offsite power circuit was >/= 521 kV.

Firm evidence exists that the condition has existed for a condition longer than allowed by TS LCO 3.6.1.3. Since the offsite power circuit voltages exceeded >/= 521 kV for more than one day prior to discovery, the TS LCO 3.6.1.3 Completion Time of four hours was exceeded; therefore this was considered a condition prohibited by Technical Specifications and is REPORTABLE as an LER.

Since the valve failed to close, there were times when the HPCS pump required flowrate would be less than the Technical Specification Surveillance Requirement (SR) 3.5.1.4 flow rate of 7115 gpm at head of >/= 445 psid. Using the data noted above, a period of > 14 consecutive days was searched for periods when the offsite power circuit voltage exceeded >/= 521 kV. This period was chosen because it would have exceeded TS LCO 3.5.1 Required Action B.2 Completion Time of 14 days. Based on review of the data between 2005 through 2008 there were no periods that the offsite power circuit voltage was >/= 521 kV that exceeded 14 consecutive days, therefore there was no TS non-compliance with TS LCO 3.5.1 Required Action Completion Times and therefore no reportability concern exists in regard to TS 3.5.1.

D. CAUSE of OCCURRENCE

The cause of valve 1E22-F012 to fail to close is due to its supply breaker instantaneous over current trip settings being set too low. It was discovered that when the offsite power circuit voltage is >/= 521 kV, the power supply breaker for valve 1E22-F012 could trip on over current, thus rendering the valve inoperable. This then could result in valve 1E22-F012 to fail to be closed from the control room.

The valve supply breaker tripped because the valve motor's instantaneous current (inrush current) exceeded the breaker instantaneous setting for an adequate duration to actuate the instantaneous over current trip. The valve motor's instantaneous current exceeded the breakers instantaneous over current trip settings because the motor experienced a hard reversal in the CLOSED direction when the valve was traveling in the OPEN direction causing higher than normal inrush currents. The valve experienced a hard reversal in the CLOSED direction because the valve's logic has the valve CLOSE when the system pressure is low or the system discharge flow is high. This hard reversal condition causes the inrush currents to be higher than normal inrush currents especially since this motor has a lot of inertia because it is a two pole motor and operates at 3600 rpm. Normal inrush currents are defined as currents drawn by a motor that starts from a stopped position.

Troubleshooting efforts after the valve supply breaker trip showed that the hard reversal inrush currents (70 to 80 amps) were actually higher than the breaker instantaneous over current trip setting (68 amps). On the day of the trip the grid voltage was higher than normal (510 to 515 kV) at 521KV. High grid voltage contributed to high buss voltages, which result in higher voltages at the supply breaker for 1E22-F012. The buss, supplying power to the valve supply breaker for 1E22-F012 had been adjusted previously, in the early 1990s, to a higher voltage setting to account for under voltage events. Therefore, high grid voltages, combined with increased voltage setting on the buss, resulted in higher instantaneous currents for the 1E22-F012 valve motor actuator, thus resulting in tripping open of the power supply breaker to the valve.

E. CORRECTIVE ACTIONS

Immediate Corrective Actions – The following actions were addressed in Condition Report GGN-2008-1201:

1. Engineering Change #6418 and Work Order #142136 were issued and completed by March 07, 2008 which replaced the valve 1E22-F012 supply breaker and increased the instantaneous over current trip settings to ensure the breaker would not inadvertently trip with offsite power circuit voltages within design specifications.

2. Breaker settings for the other ECCS motor operator valves (MOV), that were susceptible to this condition, were checked and found to be acceptable.

Long Term Corrective Actions - Condition Report GGN-2008-1201 will address any additional long-term corrective actions.

F. SAFETY ASSESSMENT

The 1E22-F012 low flow valve safety functions are containment isolation and to prevent the HPCS pump from overheating. Additionally, the valve can affect the HPCS pump safety function if the valve does not close by affecting pump flow into the reactor vessel. In this condition, the flow for the HPCS pump would be less than the Technical Specification flow rate of 7115 gpm at head of >/= 445 psid as required by SR 3.5.1.4 and the design required flow rate. The discussion provided below evaluates the safety functions.

Containment Isolation Function: For the containment isolation function, the function was maintained since the manual isolation function was maintained. There is indication in the control room that would clearly provide indication that the valve is in a position that needs to be corrected. Four hours is an adequate amount of time for operations personnel to close the valve, therefore there was no loss of safety function for the PCIV.

HPCS Pump Function – Overheating: For the valve open function, a low flow bypass line with a motor-operated gate valve connects to the HPCS discharge line upstream of the check valve on the pump discharge line. The low flow line opens and bypasses water to the suppression pool to prevent pump damage due to overheating when other discharge line valves are closed. The valve automatically closes when flow in the main discharge line is sufficient to provide required pump cooling. If the valve failed to close, there is no risk of the pump overheating. If the valve failed in the mid-position, then adequate minimum flow could not be assured to prevent pump damage during testing. However, pump overheating due to lack of minimum flow would not be a concern during LOCA (loss of coolant accident) conditions since minimum flow would be directed to the reactor.

HPCS Flow to the Reactor Vessel – Reduced Flow: With the valve 1E22-F012 assumed to fail in the full open position, there were periods of time when the HPCS pump flow delivered to the reactor vessel would be less than the Technical Specification SR 3.5.1.4 required flow rate of 7115 gpm at >/= 445 psid and less than the minimum design flow rate. However, the ECCS performance (LOCA) analyses uses relaxed ECCS parameters that are bounded by (more conservative than) the design parameters. The HPCS design flow rates and the analytical (relaxed) flow rates used in the ECCS (LOCA) analyses, together with the calculated flow rate to the reactor vessel assuming the minimum flow valve is full open, are as follows:

			HPCS Flow – Low Flow Valve Open See Note 2	
Pressure Differential (psid) (vessel to suction)	**Required Flow (gpm) (Design)** See Note 1	**Relaxed (Analytical) Flow (gpm)** See Note 3	**Clean Pipe/Normal Suppression Pool Water Level (gpm)**	**Fouled Pipe/Minimum Suppression Pool Water Level (gpm)**
1177	550	495	1608	1191
1147	1650	1485	1858	1473
200	7000	6300	6765	6728

Notes

1. Design flow rates from GGNS UFSAR Table 6.3-2.
2. Calculated flow rates are from Reference 1. The roughness value used for “fouled pipe” was that of concrete (0.001), which is 6.7 times the roughness value for commercial clean piping (.00015) and is therefore a very conservative roughness to use for calculation of flow with “fouled piping”.
3. Relaxed HPCS flow rates assumed in the LOCA analyses (Reference 2, 3, 4).

F. SAFETY ASSESSMENT (cont.)

HPCS flow is determined from the relaxed flow characteristic curve using linear interpolation as shown in Figure 1. Note that, as discussed in UFSAR (Updated Final Safety Analysis Report) Section 6.3.2.2.1, this curve is conservative relative to the HPCS design curve given in UFSAR Table 6.3-2. The information in the above table shows that the relaxed HPCS flow rates are generally bounded by (lower than) the minimum calculated flow rates with the low flow valve open (fouled pipe/minimum suppression pool level). The only shortfall is over a limited differential pressure range near 1147 psid (see Figure 1). The maximum shortfall is 12 gpm at 1147 psid (1473 gpm limiting calculated flow rate versus 1485 gpm used in the LOCA analysis). This small difference has no impact on the results of the limiting LOCA events (Design Basis Accidents) or with compliance with 10CFR50.46 acceptance criteria as discussed below.

A review of the LOCA break spectrum analyses in place during the past three years (References 2 through 4) demonstrates that the limiting (highest peak clad temperature) breaks are the large (double-ended guillotine) reactor recirculation line breaks. For example, the current cycle (Cycle 16) limiting LOCA is the double-ended guillotine reactor recirculation line break with Low Pressure Coolant Injection (LPCI) Diesel Generator (DG) single failure (SF-LPCI/DG). The PCT (peak clad temperature) for this case is 1895 degrees F (Reference 2 and UFSAR Table 6.3-3). For the large breaks, vessel pressure decreases rapidly as stated in UFSAR Figure 6.3-18A. No credit for ECCS flow is assumed until ECCS valves are fully open and ECCS pumps have reached rated speed. For the current cycle limiting LOCA, HPCS flow does not begin until 41 seconds as stated in UFSAR Table 6.3-1. At this time, vessel pressure has decreased to < 600 psia as shown in UFSAR Figure 6.3-18A. At this low pressure, the minimum calculated HPCS flow (assuming HPCS low flow valve open) is greater than the analytical flow rate. This is illustrated in Figure 1, which compares the analytical HPCS flow rate to the worst-case flow with the low flow bypass valve open. A review of other current cycle large break LOCA calculations (Reference 5) shows that, for large break areas less than the limiting break area, a similar result holds (vessel pressure is below the limiting point at the time of HPCS injection). As such, the limiting large break LOCA analyses remain bounding and the PCT does not increase due to this condition.

Recirculation line breaks between 0.1 and 1.4 square feet are analyzed for small breaks. As discussed in References 3 and 4, small breaks are significantly bounded (in terms of PCT) by the larger break LOCAs. The small (<1%) reduction in HPCS flow over the limited pressure range (approximately 1147 psid) does not impact the small break analyses and will not cause these smaller break LOCAs to become limiting. For the current cycle, the limiting small breaks are cases with single failure LPCI/DG (Reference 3). A review of the small break LOCA calculations with SF-LPCI/DG (Reference 5) shows that the PCT excursion occurs after vessel depressurization (Automatic Depressurization actuation) when reactor pressures are below approximately 800 psia. At these low pressures, the minimum calculated HPCS flow (assuming HPCS low flow valve open) is greater than the analytical flow rate defined by the above curve. Prior to vessel depressurization, the very small (<1%) reduction in available HPCS flow at approximately 1147 psid would not affect the analyses. As such, the PCT excursions for the small break cases are not impacted by the small HPCS flow reduction at high vessel pressure. In addition, for previous cycles, the limiting (highest PCT) small breaks are cases with HPCS single failure (Reference 4). These cases do not credit any flow from the HPCS system and are thus unaffected by the condition.

The limiting non-recirculation (ECCS) line breaks for both the current and previous cycle analyses are the HPCS line breaks. These cases do not credit any flow from the HPCS system and are thus unaffected by the condition (Reference 3, 4). ECCS line breaks have similar characteristics to a small recirculation line break LOCA. Similar to the small recirculation line break cases described above, the small reduction in HPCS flow

F. SAFETY ASSESSMENT (cont.)

has no impact on the small break analyses and would thus not result in the other ECCS line breaks (LPCI, LPCS) becoming more limiting.

Based on the above discussion, the HPCS system would still perform its design basis safety function to prevent the peak fuel cladding temperature from exceeding the 10CFR50.46 acceptance limit during a LOCA with the low flow valve in the open position. This condition did not prevent the fulfillment of a safety function and there were therefore no safety system functional failures. As such, the health and safety of the public was not compromised by this event.

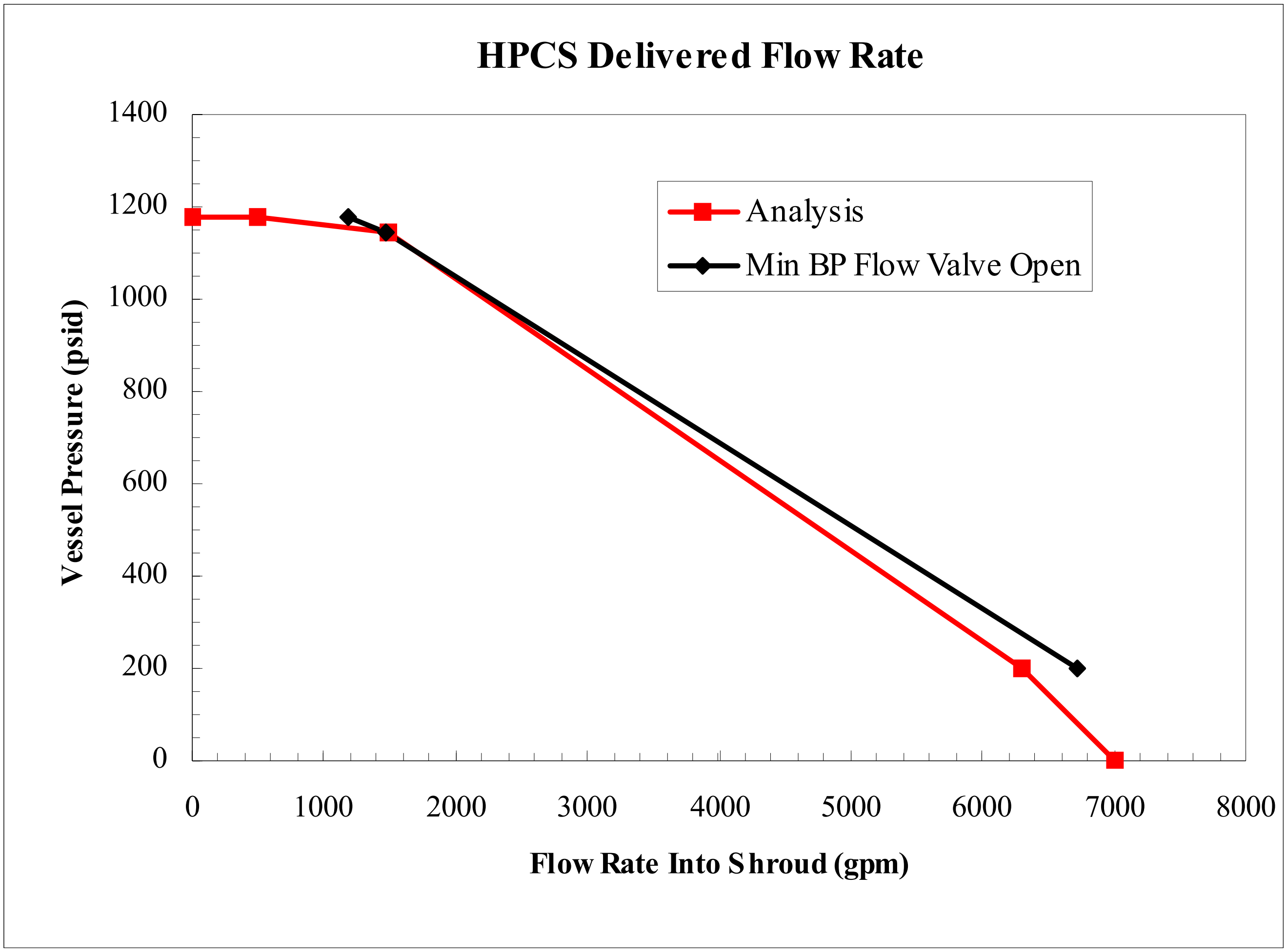
References:

1. GGNS Calculation MC-Q1E22-92002, Rev. 0.
2. EMF-3177(P), Rev. 2, "Grand Gulf Nuclear Station EXEM BWR-2000 LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM-10 Fuel," December, 2006 (UFSAR Section 6.3.6 Ref. 11).
3. EMF-3176(P), Rev. 1, "Grand Gulf Nuclear Station EXEM BWR-2000 LOCA Break Spectrum Analysis for ATRIUM-10 Fuel," July, 2005 (UFSAR Section 6.3.6 Ref. 12).
4. EMF-2539(P), Rev. 0, "Grand Gulf Nuclear Station LOCA Break Spectrum Analysis," April, 2001.
5. AREVA calculation E-7193-S07-12.

G. ADDITIONAL INFORMATION

Previous Similar Events - Pursuant to 10CFR50.73(b)(5) this issue is considered an infrequent event. There has not been any occurrence of the same underlying concern in the past two years at Grand Gulf Nuclear Station. There was a similar occurrence at Clinton Power Station as documented in LER 1999-012-000 dated December 20, 1999.

Figure 1


HPCS Delivered Flow Rate
Analysis
Min BP Flow Valve Open
Vessel Pressure (psid)
Flow Rate Into Shroud (gpm)
0
200
400
600
800
1000
1200
1400
1000
2000
3000
4000
5000
6000
7000
8000