GPU Nuclear

GPU Nuclear Corporation
Route 441 South
P.O. Box 480
Middletown, Pennsylvania 17057-0480
(717) 944-7621
Writer's Direct Dial Number:

May 23, 1996
6710 -96-2136

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Inservice Testing of ASME Code Class 1, 2, and 3 Pumps and Valves (IST) - Response to NRC Questions Regarding Code Relief for the Third Ten Year Interval

GPU Nuclear submitted the TMI-1 IST program description for the third 10-year IST interval on September 21, 1995. Included were requests for relief from Code requirements which include 1) Code requirements that have been determined to be impractical for TMI-1 in accordance with 10 CFR 50.55a(f)(5)(iii) and 2) alternatives proposed in accordance with §50.55a(a)(3).

The NRC's comments on our relief requests were discussed in several conference calls which included Mr. Joseph Colaccino of NRR. The attachment includes a revision or restatement of the original relief requests followed by the NRC comment and the GPU Nuclear response. Revised test appears in bold; deleted text appears with a line through it.

Sincerely,

MJ Ross for

J. Knubel
Vice President and Director, TMI

Enclosure

cc: Administrator, Region I
TMI-1 Senior Project Manager
TMI Senior Resident Inspector

A047

RELIEF REQUEST NO. P 1 (REVISED)

Tag No.	Component	Type
MU-P1A	Makeup & Purification Pump "A"	Centrifugal
MU-P1B	Makeup & Purification Pump "B"	Centrifugal
MU-P1C	Makeup & Purification Pump "C"	Centrifugal

Code Section from Which Relief is Requested

1. **OM-6, §5.6, "Duration of Tests," requirement for a run time of at least two minutes after reaching stable pump conditions before obtaining data, and**
2. **OM-6, §5.2, "Test Procedure," requirement for testing at a single reference point.**
3. ~~**OM-6, §4.6.4, vibration measurement requirements.**~~

Alternate Test Description

As permitted by NRC GL 89-04, Position No. 9, the pumps will be full flow tested each refueling outage (see justification). The refueling outage test will include measurement of stable flow rate, differential pressure, and vibration. Pump testing will be performed with the system lined up to pump to the RCS through different flow path combinations to provide pump data at various flowrates. Run time through each flow configuration may be less than the two minutes required by OM-6. ~~Due to the short duration of testing, a best effort will be used to take vibration data and all points may not be obtained.~~

Basis for Relief Request

The amount of time that the Makeup Pump injects at full flow to the RCS must be limited. Pumping to the RCS will raise pressurizer level and a plant transient can occur. Run time therefore must be limited.

Pumping time is limited to a total of approximately 5 min for all flow configurations. Because of the short time available for a test run, throttling to a specific reference point can not be accomplished. The pump is run with several different valve lineups to verify that flowrate and head are equal to or higher than accident design requirements. Flow rate and pressure measurements for each lineup is compared with previous test data. Acceptance is based on meeting or exceeding accident flow and head requirements. This meets the intent of the code. The test is similar to that described in NUREG 1482, §5.2 except for the following:

1) A manufacturers curve is not used. Comparison is with the FSAR Safety Analysis curve and previous full flow tests.

2) A five point curve is not used. The pump will operate at several different points~~, and~~

3) ~~Vibration is taken during the quarterly test and to the extent time allows during the full flow test~~.

These tests demonstrate pump operability and meet the intent of the code.

NRC Comments on Relief Request P1 and GPU Nuclear Response:

NRC Comment 1:
Has there been any maintenance performed on any of the three pumps?

GPU Nuclear Response:
In addition to routine maintenance, GPU Nuclear has installed balance discs on the inboard and outboard bearing of each pump which has improved the vibration levels.

NRC Comment 2:
Has the FSAR curve been validated?

GPU Nuclear Response:
Yes the FSAR curves have been validated.

NRC Comment 3:
All vibration data, including data in each direction during full flow testing, must be collected. Licensee's proposed testing is unacceptable.

GPU Nuclear Response:
GPU Nuclear has decided to withdraw the request for relief from the requirements to take all of the vibration data.

RELIEF REQUEST NO. P 2 (REVISED)

Tag No.	Component	Type
EF-P1	Turbine Driven EF Pump	Centrifugal
EF-P2A	Motor Driven EF Pump "A"	Centrifugal
EF-P2B	Motor Driven EF Pump "B"	Centrifugal

Code Section from Which Relief is Requested

Regarding refueling interval tests, relief is requested from:

1. OM-6, §5.6, "Duration of Tests," requirement for a run time of at least 2 minutes after reaching stable pump conditions before obtaining data,
2. OM-6, §5.2.c, requirement to compare flow rate and pressure to their respective reference values~~, and~~
3. ~~OM-6, §4.6.4.a, requirement to take vibration measurements on each accessible pump bearing~~.

Alternate Test Description

TMI-1 Tech Specs requires a test each refueling to demonstrate EFW pumps can pump water from the Condensate Storage Tanks (CSTs) to the Once Through Steam Generators (OTSGs). ~~Except for tests every 3rd refueling outage, the pump will be stopped as soon as accident design flowrate is achieved.~~

~~Every 3rd refueling (starting 11R in 9/95), the test will demonstrate full flow for each pump based on a reference value. During the full flow test, the pump will be stopped based on OTSG level when ~1000 gallons has been transferred.~~ Run time while pumping to the OTSG ~~in either case~~ may be less than two minutes as required by OM-6. ~~Best effort will be made to take vibration data for each pump bearing, but because of the short duration of the refueling test, it may not be possible to obtain all data.~~

During the next refueling outage (12R), GPU Nuclear will verify the pump curve to be the valid accident design curve in accordance with the guidance in NUREG-1482, §5.2, except that only three points on the curve will be taken. Thereafter, each refueling outage, a full flow test of all three pumps will be performed ~~every 3rd refueling outage~~ in which accident design flow **and differential pressure** for each pump will be verified by running the pumps only long enough to take **stable flow, differential pressure, and vibration data; pressure and flow will be compared to the reference curve. This test verifies acceptable flow rate and differential pressure of the pumps.** ~~In between full flow tests, the refueling interval test will demonstrate accident design flowrate.~~

Basis for Relief Request

The EFW pumps are only used for emergency operation. They are not used for startup, shutdown, or normal plant operation. EFW flow to the OTSGs is limited by the cavitating venturis. Since the pumps operate only for test, no significant degradation is expected.

Since the refueling interval tests transfer lower quality water to the OTSGs, the number and duration of tests must be limited to minimize routine exposure of the OTSGs to lower quality water. Minimizing test duration is necessary to limit the amount of water injected into the OTSGs where corrosion damage promoted by O_2 can occur. **Throttling to a reference value of flow/differential pressure as well as waiting out a minimum of two minutes run time lengthens the amount of time the EFW pumps are running while pumping oxygenated water into the OTSGs.**

Justification for our request is based on limiting the amount of oxygenated water that is pumped into the OTSGs to minimize the potential for steam generator tube degradation. The feedwater oxygen concentration limit for normal plant operation is 5 ppb; and TMI-1 normally maintains feedwater oxygen concentration less than 1 ppb. At the lower temperatures during shutdown, hydrazine is unable to react and scavage the oxygen. So to help lower the oxygen content, prior to the test a nitrogen blanket is applied to the CSTs through spargers. We can typically lower the oxygen concentration of the water to approximately 200 - 300 ppb. This range is still many times higher than the limit recommended by The Electric Power Research Institute (EPRI)[1]. We believe that performing a shorter test will not compromise our ability to adequately demonstrate EFW pump operability and meets the intent of the Code.

~~Because of the short test duration, it may not be possible to obtain all vibration data.~~

EFW pump quarterly tests verify the pumps are operational, start on demand, and generate the required discharge pressure. During the quarterly test, vibration data will be taken on each bearing while pumping through the recirculation line.

NRC Comments on Relief Request P2 and GPU Nuclear Response:

NRC Comment 1:
Test frequency and methodology are unacceptable as proposed. No basis for not full flow testing each pump every refueling outage. Unsure of how other B&W plants perform this testing. Perhaps licensee should make inquiries.

GPU Nuclear Response:
GPU Nuclear contacted the other B&W plants and found that others, some of which have full flow test loops, are performing a refueling interval test as required by the code. We are withdrawing our request to take less vibration data than that required by the code. This reduces the extent of the relief requested to the issue of run time and throttling to a reference value. We are requesting relief to limit the quantity of oxygenated water that is pumped into the OTSGs. This will minimize the potential for steam generator tube degradation.

NRC Comment 2:
All vibration data must be collected during pump testing.

GPU Nuclear Response:
GPU Nuclear withdraws the portion of this request regarding relief from the requirements to take all of the vibration data.

---

[1] EPRI TR-102134, Final Report, "PWR Secondary Water Chemistry Guidelines - Revision 3," May 1993.

RELIEF REQUEST NO. P 3 (REVISED)

Tag No.	Component	Type
NR-P1A	Nuclear Service River Water Pump "A"	Centrifugal
NR-P1B	Nuclear Service River Water Pump "B"	Centrifugal
NR-P1C	Nuclear Service River Water Pump "C"	Centrifugal

Code Section from Which Relief is Requested
Relief is requested from OM-6, §5.2, "Test Procedure," item (d) regarding the determination of flow rate.

Alternate Test Description
Flowrate for individual pumps will be measured at refueling outages.

Basis for Relief Request
The test flow instrumentation for this system is located in the common discharge from all three pumps. The piping configuration does not facilitate installation of individual pump flow measuring devices. GPU Nuclear has not been successful in attaining acceptable accuracy or repeatability using individual annubar flow instruments.

To read total NR pump flow (NR-FI-290), for any pump combination including two pump operation, it is necessary to isolate makeup water to the circulating water flume by closing the 30" butterfly valves (NR-V4A and NR-V4B). This directs all NR Pump flow to the Nuclear Service Heat Exchangers and reduces the temperature of Nuclear Services Closed Cooling Water which cools the Reactor Coolant Pump (RCP) seal return coolers. This results in a decrease in Makeup System supply water temperature including the supply for Reactor Coolant Pump Seal water. The resulting fluctuations in Reactor Coolant Pump Seal leakoff flow rate reduces the performance of the pump seals and adds to the risk of eventual RCP seal damage.

During normal plant operation, at least two of the three pumps are in operation. Operation of only one Nuclear Service River Water Pump ~~is not allowed because of reliability concerns and~~ could jeopardize plant equipment due to **system** heat loads ~~for a large part of the year~~. **If all but one NR Pump were secured for the purpose of testing, this would cause significant temperature variations in safety related components, add significant operator burden to assure adequate cooling of the many plant components that would be affected, and result in some operational risk.**

Individual NR Pump flow rate measurement is impractical during plant operation or during Cold Shutdowns of short duration. **The quarterly test to measure differential pressure and vibration as well as the refueling test to verify head and flow rate greater than the accident design will continue to assure operability of the NR Pumps.**

NRC Comments on Relief Request P3 and GPU Nuclear Response:

NRC Comment:
There is not enough information to evaluate this relief request. The licensee has not addressed the effect of measuring flow rates for two pumps will have on their Code acceptance criteria.

GPU Nuclear Response:
Response to this comment is provided in the revised text of the relief request.

RELIEF REQUEST NO. P 4 (NEW RELIEF REQUEST[2])

Tag No.	Component	Type
DH-P1B	Decay Heat Pump "B"	Centrifugal

Code Section from which Relief is Requested
Relief is requested from OM-6, §6.1, "Acceptance Criteria" which requires doubling the test frequency until the cause is determined and corrected for vibration readings in the alert range (0.325 ips).

Alternate Test
The alert range for DH-P1B will be raised to vibrations greater than 0.400 ips (vs greater than 0.325 ips)in the vertical direction. The alert range for the horizontal direction will remain at vibration levels greater than 0.325 ips.

Basis for Relief
OM-6 requires doubling test frequency when the overall vibration amplitude is greater than 0.325 ips during quarterly testing. The Code assumes that the equipment has degraded to the point where more frequent monitoring and possibly a repair are warranted. There is no consideration for test conditions, vibration history, or equipment maintenance history. These pumps are tested each refueling at both medium and high flow rates where the vibration levels have always been lower with the majority of vibration occurring at the vane pass frequency.

Consideration of vibration amplitudes was not part of the original acceptance criteria for many of the pumps procured for earlier nuclear plants. As a result, some pumps purchased in the late 1960's and early 1970's had inherently high vibrations. During low flow conditions, typical of IST testing, vibration amplitudes are at their highest. TMI plant data, shop testing of the spare pump by GPUN, and conversations with several pump vendors indicate that it is not unusual to experience vibrations in excess of 0.325 ips with the older pumps, especially at low flow conditions. Provided there is a successful long term operating history and provided there is no significant change in vibration amplitude or spectra, there is no reason to suspect equipment degradation at these vibration levels. TMI's DHR pumps are one specific example and show the type of evaluations that we perform for pumps that exceed the alert limit.

TMI's DHR pumps are early edition API 610 process pumps. They have operated with occasionally high vibration since 1974. This includes extensive operating time between 1979 and 1985 (TMI's extended shutdown which lasted approximately 6 1/2 years). The pumps have not failed, there is no unusual degradation in hydraulic performance, and seal and bearing life are normal. Vibration amplitudes average 0.293 ips (standard deviation of 0.100) with the highest vibration occurring at the lower flow IST conditions. Because of normal variation in vibration response and measurement, measured vibration exceeds 0.325 ips about once per year during low flow IST operation. However, there is no upward trend in the data and the greatest majority of vibration has always been at vane pass frequency.

GPU has discussed these relatively high vibration readings with several vendors who manufactured API pumps. The vendors stated that high vibrations are expected with early edition API 610 pumps,

---

[2] This new specific relief request has been added as discussed in the response to NRC comments on generic Relief Request No. PG 1 which has been withdrawn.

particularly at the low flow rates encountered during inservice testing. Additionally, GPU Nuclear has shop tested TMI's spare DH Pump and found the vibration readings were almost identical to TMI's two inservice pumps. During the shop test, vibration data were recorded at many different flow rates. At flow rates equal to and below the IST flow rate, vibrations occasionally exceeded 0.325 ips. This pump was inspected prior to and after the shop test to assure no degradation had occurred. The spare pump is the same model number; it was purchased around the same time as the inservice pumps and has never been used.

Figures 1 and 2 show vibration data from tests of the DH Pumps (DH-P1A and DH-P1B, respectively) since 1987. Points shown represent the highest vertical overall amplitude since amplitudes in the vertical direction are higher than the horizontal vibration amplitudes. The vibration spectrum is essentially all at vane pass frequency with no 1X or 2X harmonics that could indicate pump problems. These data present a clear case for raising the allowable vibration levels for DH-P1B. Several attempts have been made to reduce the vibration levels for the Decay Heat Pumps, including pump motor alignment, strengthening the backfoot, and adding lead weights to the back end of the pump.

Based on the successful operating history of DH-P1B, no step changes or trends in vibration data as shown on Figure 1, extensive vibration analysis, shop testing, and vendor input, GPU Nuclear does not consider the vibration amplitudes of DH-P1B unacceptable. Replacing or modifying the pumps to reduce vibrations only to assure they do not occasionally exceed 0.325 ips could be unnecessary. Further, doubling the test frequency would result in running the pumps more often at low flow conditions and would provide no useful information. Therefore, this request to allow raising the alert range to greater than 0.400 ips (vs greater than 0.325 ips) is justified for DH-P1B.

RELIEF REQUEST NO. PG 1 (WITHDRAWN)

~~Code Section from which Relief is Requested~~
~~Relief is requested from OM 6, §6.1, "Acceptance Criteria" which requires doubling the test frequency until the cause is determined and corrected for vibration readings in the alert range (0.325 ips).~~

~~Alternate Test~~
~~If vibration readings are in the alert range, vibration trends and spectrum data will be evaluated. If we conclude that the vibration amplitude is above the established average by a statistically significant amount or that a significant change in the spectrum has occurred, then test frequency will be doubled until the cause is determined and corrected. Otherwise normal monitoring will continue.~~

~~Basis for Relief~~
~~OM 6 requires doubling test frequency when the overall vibration amplitude reaches 0.325 ips during quarterly testing. Code assumes that the equipment has degraded to the point where more frequent monitoring and possibly a repair are warranted. There is no consideration for test conditions, vibration history, or equipment maintenance history.~~

~~Consideration of vibration amplitudes was not part of the original acceptance criteria for many of the pumps procured for earlier nuclear plants. As a result, some pumps purchased in the late 60's and early 70's had inherently high vibrations. During low flow conditions, typical of IST testing, vibration amplitudes are at their highest. TMI plant data, shop testing by GPUN, and conversations with several pump vendors indicate that it is not unusual to experience vibrations in excess of 0.325 ips with the older pumps, especially at low flow conditions. Provided there is a successful long term operating history and provided there is no significant change in vibration amplitude or spectra, there is no reason to suspect equipment degradation at these vibration levels. TMI's DHR pumps are one specific example and show the type of evaluations that we perform for pumps that exceed the alert limit.~~

~~EXAMPLE:~~
~~TMI's DHR pumps are early edition API 610 process pumps. They have operated with occasionally high but untrending vibration since 1974. This includes extensive operating time between 1979 and 1985 (TMI's extended shutdown). The pumps have not failed, there is no unusual degradation in hydraulic performance, and seal and bearing life are normal. Vibration amplitudes average 0.293 ips (standard deviation of 0.1) with the highest vibration occurring at the lower flow IST conditions. Because of normal variation in vibration response and measurement, measured vibration exceeds 0.325 ips about once per year during low flow IST operation. However, there is no upward trend in the data and vibration has always been at vane pass frequency.~~

~~GPU has discussed these relatively high vibration readings with several vendors who manufactured API pumps. The vendors stated that high vibrations are expected with early edition API 610 pumps, particularly at the low flow rates encountered during inservice testing.~~

~~Additionally, GPU has shop tested TMI's spare DHR pump and found its vibration readings almost identical to TMI's two inservice pumps. During the shop test, vibration data were recorded at many different flow rates. At flow rates equal to and below the IST flow rate, vibrations occasionally exceeded 0.325 ips. This pump was inspected prior to and after the shop test to assure no degradation had occurred. The spare pump is identical to the inservice pumps and has never been used.~~

~~Based on the successful operating history, no step changes or trends in vibration data, extensive~~

~~vibration analysis, shop testing, and vendor input, GPU Nuclear does not consider the vibration amplitudes of TMI's operating DH pumps unacceptable. Replacing or modifying the pumps to reduce vibrations only to assure they do not occasionally exceed 0.325 ips could be unnecessary. Further, doubling the test frequency would result in running the pumps more often at low flow/high vibration conditions and would provide no useful information.~~

NRC Comments on Relief Request PG1 and GPU Nuclear Response:

NRC Comment:
Relief requests to raise the vibration alert range should be done on a pump specific basis. The relief request(s) will be evaluated for each individual pump. Information on pump vibration history (including, pump, specific bearing, vibration direction and dates) can be provided in either raw or graphical form. A discussion of the investigation into the high pump bearing vibration levels, including any discussion with the manufacturer and any actions taken, should be included.

A number of these relief requests have been granted. Typically, the alert range will be increased in the specific direction to a level that exceeds historical vibration levels for that particular pump.

GPU Nuclear Response:
GPU Nuclear will limit the request at this time to Decay Heat Pump 1B (DH-P1B). This generic relief request is therefore withdrawn and a new specific relief request No. P 4 has been added for raising the vibration alert range for DH-P1B from 0.325 ips to 0.400 ips in the vertical direction. See New Relief Request No. P 4.

RELIEF REQUEST NO. PG 2 (WITHDRAWN)

~~Code Section from Which Relief is Requested~~
~~Additional clarification to OM 6, §6.1, "Acceptance Criteria," regarding rerunning of a test is requested.~~

~~Alternate Test Description~~
~~If the test parameter values fall outside the acceptable range in OM 6 due to an identified systematic error, such as an improper valve line up or inaccurate instrumentation, the test will be rerun after the correction of the error.~~

~~Basis for Relief Request~~
~~There can be instances where the data gathered during a test appears to be in question. The clarification to OM 6, §6.1 will permit the evaluation of the condition and the rerunning of the test without declaring a pump to be either in the Alert Range or inoperable. The evaluation will be included in the test records. This is permitted by OM Code ISTB 1995, §6.2.3.~~

NRC Comments on Relief Request PG2 and GPU Nuclear Response:

NRC Comment:
Licensee not allowed to request relief to use future editions of the Code without including related requirements.

GPU Nuclear Response:
This relief request is being withdraw at this time. GPU Nuclear understands that this request is premature pending generic action by ASME and NRC to approve pending code.

RELIEF REQUEST NO. PG 3 (REVISED)

Code Section from Which Relief is Requested
Relief is requested from OM-6, §6.1, "Acceptance Criteria," regarding doubling of test frequency in the alert range, and declaring a pump inoperable if vibration, flow, or differential pressure is in the alert or required action range.

Alternate Test Description
In lieu of the requirements in paragraph 6.1, "Acceptance Criteria," in OM-6 for pumps whose hydraulic and/or vibration data falls into the required action range of Table 3 "Ranges for Test Parameters," the requirements of ISTB 6.2.2, "Action Range," of the of the 1995 Edition of the ASME OM Code will be implemented for the inservice testing of safety related pumps. The related requirements of ISTB 4.6, "New Reference Values," will also be implemented for the inservice testing of safety related pumps.

If measured pump parameters fall within the alert range, the test frequency will be doubled until the cause of the deviation is found and corrected or an analysis of the pump is performed and new reference values established. If measured test parameters fall within the required action range, the pump shall be declared inoperable until either the cause of the deviation is determined and the condition corrected or an analysis of the pump is performed and new reference values established.

The analysis will include a comparison of the test results with the required design parameters, an evaluation of previous data to establish a trend, an investigation into the reason for the parameter change, and if necessary the collection of additional data. To be successful, the evaluation must conclude that the condition does not impair the ability of the pump to perform its safety function. The evaluation will be maintained in the test records.

Basis for Relief Request
An analysis of the pump condition can demonstrate that the pump can furnish its design function especially for those pumps with large margins above their design requirement. Doubling test frequency for pumps is only intended to get more data. For pumps that are normally standby, the degrading mechanism **should** not be active when the pump is off. Doubling test frequency may not establish any additional information.

Based on a successful operating history with no step changes or abnormal trends, data which may occasionally fall in the alert range may not be unacceptable. Modifying or replacing the pumps to reduce vibrations only to assure that they do not occasionally fall in the alert range could be unnecessary. Further, doubling the test frequency would result in running the pumps more often at low flow conditions and would provide no useful information.

The deviation of pump hydraulic and/or vibration data which falls in the required action range of Table 3, "Ranges for Test Parameters," (consisting of Figure 1 and Tables 3a and 3b of the errata to OMa-1988 contained in OMb-1989) is an indication of degradation of pump performance. However, such a deviation does not address the ability of the pump to perform its intended safety function or the rate of degradation. It may be possible through analysis of past performance data from the subject pump or other similar pumps to ensure that the pump remains capable of performing its intended safety function until the next scheduled surveillance test. Such an analysis could prevent the unnecessary shutdown of the unit to perform repairs to the pump.

These provisions were permitted by earlier editions of ASME Section XI, Subsection IWP, and are currently permitted by the ASME OM Code ISTB-1995, §§6.2.2 **and 4.6**.

NRC Comments on Relief Request PG3 and GPU Nuclear Response:

NRC Comment:
This can be done per 10 CFR 50.59. Similar relief requests have been denied. See also PG2.

GPU Nuclear Response:
GPU Nuclear has learned that similar relief has been granted other licensees. Having contacting another licensee with a similar request, the above has been revised to incorporate the intent of a request which has been approved by the NRC.

RELIEF REQUEST NO. V 1 (WITHDRAWN)

~~Tag No.~~	~~Component~~	~~Category~~	~~Type~~	~~Actuator~~
~~MU V78~~	~~RCS Fill Isol Vlv Associated with Makeup Tank and Pumps Bypass~~	~~C~~	~~Globe~~	~~Hand~~
~~MU V79~~	~~RCS Fill Chk Vlv Associated with Makeup Tank and Pumps Bypass~~	~~C~~	~~Sw Chk~~	~~NA~~

~~Code Section from Which Relief is Requested~~
~~Relief is requested from the exercising requirements of OM 10, §4.3.2.2.~~

~~Alternate Test Description~~
~~The valve pair (MU V78 in series with MU V79) is verified closed by normal operating procedures. No specific test procedures will be implemented. MU V78 is a locked closed hand operated globe valve. Leakage will be noted by observing Makeup Tank level and pressure as part of normal operating procedures.~~

~~Basis for Relief Request~~
~~Relief is requested from the requirement to perform a quarterly exercise test for MU V79. MU V79, a 2½" swing check valve, has no nuclear safety function in the open position. During normal plant operation, MU V79 is isolated by a hand operated locked closed valve, MU V78. Either of these valves fulfills the nuclear safety function. MU V78 is maintained locked closed during operation in accordance with the normal system valve lineup. Both of these valves are Seismic Category I and capable of seating against the shutoff head of the Makeup Pumps.~~

~~A small amount of leakage past MU V78 and MU V79 into the 2½" fill line is of no consequence since any leakage water would return to the Makeup Tank and remain available for makeup to the primary system. The Makeup Tank is alarmed for both pressure and level.~~

~~Significant leakage such as would be seen if both valves were open would be identified by significant changes in Makeup Tank pressure and level. Significant leakage would also be identified by a refueling surveillance 1303-11.8, "HPI Test," since this procedure verifies HPI accident design flow rate.~~

~~NUREG 1482, §4.1.1 discusses the situation of two check valves in series and the acceptability of testing them together. Although MU V78 is not a check valve, a similar argument can be made. There is no pressure tap between these two redundant valves and testing would only be capable of verifying the seat tightness of one valve. Both of these valves are capable seating against accident design pressure. The valve pair (MU V78 and MU V79) are under constant test for significant leakage since Makeup Tank high level and high pressure are both alarmed conditions. Therefore, specific tests to exercise these valves or disassemble and inspect them are not needed.~~

NRC Comments on Relief Request V1 and GPU Nuclear Response:

NRC Comment:
Is valve MU-V79 required to be exercised in your IST program since MU-V78 is locked closed?

GPU Nuclear Response:
No, MU-V79 (a check valve is in series with hand operated locked-closed globe valve MU-V78) has no open safety function. Also, a small amount of leakage past MU-V78 and MU-V79 would be of little consequence since the leaking fluid would remain in the Makeup and Purification System. Increased leakage would be evident by an increase in Makeup Tank level. Therefore this valve does not require leak rate testing. Since the discharge pressure of the running Makeup Pump provides the closing force, MU-V79 cannot be exercised during operation. It is physically impossible to open MU-V79 during normal system operation because RCS makeup and RCP seal injection flows are necessary for steady state plant operation. Exercising MU-V78 valve would provide no meaningful information since the valve has no open safety function and MU-V79 can not physically open or be opened during normal plant operation. GPU Nuclear has concluded that these valves do not need to be included in the TMI IST program. Therefore, this relief request is being withdrawn.

RELIEF REQUEST NO. V 2 (WITHDRAWN)

~~Tag No.~~	~~Component~~	~~Category~~	~~Type~~	~~Actuator~~
~~DH V50~~	~~Spent Fuel Return Cleanup Chk Vlv~~	~~C~~	~~Swing Check~~	~~NA~~

~~Code Section from Which Relief is Requested~~
~~Relief is requested from the exercising requirements of OM 10, §4.3.2.2.~~

~~Alternate Test Description~~
~~The valve pair (DH V50 in series with SF V44) is verified closed by means of Borated Water Storage Tank (BWST) level as part of normal operating and alarm response procedures. No specific test procedure will be implemented.~~

~~Basis for Relief Request~~
~~DH V50 has no nuclear safety function in the open position. The nuclear safety function is to prevent fluid from the BWST and Reactor Building Sump from entering the non seismic portion of the Spent Fuel Pool Cooling System (SF) after a LOCA when the Decay Heat Removal System is in Low Pressure Injection (LPI) mode and suction is from the BWST and later from the Reactor Building sump. The valve is in series with a hand operated normally closed valve (SF V44).~~

~~The leak tightness of these valves (DH V50 and SF V44) is under constant test by surveillance of BWST level. Maintaining BWST level, which is also alarmed, verifies that (either DH V50 or SF V44) is in the closed position. Any appreciable leakage would be detected through use of the current procedures (e.g., Surveillance Procedure 1301 1, "Shift and Daily Checks"). In addition, refueling surveillance 1303 11.54, "LPI Test," verifies that these valves do not have significant seat leakage since this procedure verifies LPI accident design flow rate.~~

~~DH V50 is only used when the DH system is to be filled from the spent fuel pool which is rarely done. SF V44 is included in the Augmented IST program because it is a non Code valve. Both of these valves are Seismic Category I and capable of seating against accident design pressure. Either of these valves fulfills the nuclear safety function.~~

~~NUREG 1482, §4.1.1 discusses the situation of two check valves in series and the acceptability of testing them together. Although SF V44 is not a check valve, a similar argument can be made. There is no pressure tap between these two redundant valves and testing would only be capable of verifying the seat tightness of the first valve, DH V50. The valve pair (DH V50 and SF V44) are under constant test for significant leakage since BWST level is alarmed and is surveilled. Therefore, opening SF V44 to test the leak tightness of DH V50, or other specific tests to exercise these valves or disassemble and inspect them are not needed.~~

NRC Comments on Relief Request V2 and GPU Nuclear Response:

NRC Comment:
Is valve DH-V50 required to be exercised in your IST program?

GPU Nuclear Response:
No. DH-V50 has no nuclear safety function in the open position but has a closed safety function to prevent fluid from the Reactor Building Sump from entering the nonseismic portion of the Spent Fuel System when Decay Heat Removal is in the Low Pressure Injection (LPI) mode with suction from the Reactor Building Sump. SF-V-44, a diaphragm valve in series with DH-V-50 and outside the ISI boundary, is also normally closed. Either of these valves serves to maintain LPI system integrity.

The head from the BWST is greater than the head DH-V50 and/or SF-V44 would ex rience during accident conditions. Maintaining BWST level, which is alarmed, verifies that DH-V50 and/or SF-V44 are in the closed position and not leaking. GPU Nuclear has concluded that DH-V50 does need not need be included in the IST program. Therefore, this relief request is being withdrawn.

RELIEF REQUEST NO. V 3 (WITHDRAWN)

~~Tag No.~~	~~Component~~	~~Category~~	~~Type~~	~~Actuator~~
~~BS V52A~~	~~NAOH Tank LPI/BS Suct Hdr Chk Vlv~~	~~C~~	~~Swing Check~~	~~NA~~
~~BS V52B~~	~~NAOH Tank LPI/BS Suct Hdr Chk Vlv~~	~~C~~	~~Swing Check~~	~~NA~~

~~Code Section from Which Relief is Requested~~
~~Relief is requested from OM 10, §4.3.2.4(e) regarding the frequency of disassembly for inspection purposes. This valve cannot be exercised with flow.~~

~~Alternate Test Description~~
~~These valves form a group as described in Generic Letter 89 04, Position 2 and NUREG 1482 in that they are identical, mounted the same manner, and see the same service. As approved for the last 10 year interval, one valve in the group will be disassembled and inspected and the disc manually exercised every other refueling outage alternating between A and B, such that each valve is disassembled every fourth outage. This results in a disassembly frequency of approximately every 8 years for a specific valve. If the inspected valve is found to be unacceptable, the other valve in the group will be inspected during the same outage.~~

~~Basis for Relief Request~~
~~Flow testing of these valves requires the injection of sodium hydroxide into the Decay Heat Removal and Building Spray Systems. Such operation is impractical in that the sodium hydroxide must be removed from the Decay Heat System. Therefore, flow testing is not considered to be practical. Except for accident conditions, these valves would never see flow. The valves are stainless steel and are subject to conditions (clean, low temperature fluid) that would minimize the possibility of their sticking closed.~~

~~Both valves were disassembled and inspected in 1984. BS V52A was again disassembled during the 6R (1986) and 9R (1991) Outages. BS V52B was disassembled during the 7R (1988) Outage and in July, 1995. All inspections have shown the valves to be in excellent condition.~~

~~Given the valves' materials of construction, lack of service, and the results of previous disassembly inspections, this relief is justified. Similar relief was approved by the NRC for the previous IST interval in a letter dated December 27, 1989. The NRC granted relief based on GPU Nuclear letters dated June 7, 1988 and April 17, 1989. Inspection results from disassembly subsequent to our previous correspondence provide even greater justification for reducing the frequency of inspection.~~

NRC Comments on Relief Request V3 and GPU Nuclear Response:

NRC Comment:
Do the valves listed in these relief requests (V3, V4, and V5) meet the guidance for extending valve inspection frequency due to extreme hardship as described in GL 89-04, Position 2 and NUREG-1482, pages A-7 through A-15?

GPU Nuclear Response:
Yes. An evaluation has been prepared in accordance with NUREG 1482, "Guidelines for Inservice Testing at Nuclear Power Plants," NRC Staff Position 2, Page A-8, which states that disassemble/inspection can be longer that once every six years, if the following information is developed:

1. Disassemble and inspect each valve in the valve grouping and document in detail the condition of each valve and the valve's capability to be full-stroked.

2. A review of industry experience, for example, as documented in NPRDS regarding the same type of valve in similar service.

3. A review of the installation of each valve addressing the "EPRI Applications Guidelines for Check Valves in Nuclear Power Plants" for problematic locations.

The evaluation,[3] which is documented in the IST program, allows one valve (BS-V52A or BS-52B) to be disassembled/inspected every other refueling outage, alternating between A and B valves. Therefore, relief is not needed.

---

[3] GPU Nuclear Memorandum 3310-96-0007, dated March 13, 1996.

RELIEF REQUEST NO. V 4 (WITHDRAWN)

~~Tag No.~~	~~Component~~	~~Category~~	~~Type~~	~~Actuator~~
~~BS-V30A~~	~~Ctmt Isol - BS Nozzle Inlet Chk Vlv~~	~~C~~	~~Swing Check~~	~~NA~~
~~BS-V30B~~	~~Ctmt Isol - BS Nozzle Inlet Chk Vlv~~	~~C~~	~~Swing Check~~	~~NA~~

~~Code Section from Which Relief is Requested~~
~~Relief is requested from OM 10, §4.3.2.4(c), regarding disassembly frequency. This valve cannot be full stroke exercised with flow.~~

~~Alternate Test Description~~
~~These valves form a group as described in Generic Letter 89-04, Position 2 and NUREG 1482 in that they are identical, mounted the same manner, and see the same service. As approved for the last submittal, each valve in the group will be disassembled and inspected and the disc manually exercised every other refueling outage alternating between A and B, such that each valve is disassembled every fourth outage. This equates to disassembly frequency of approximately 8 years for a specific valve. If the inspected valve is found to be unacceptable, the other valve in the group will be inspected during the same outage. Both valves are partial flow tested every quarter using nitrogen.~~

~~Basis for Relief Request~~
~~Operation of this system would require spraying the containment. Therefore flow testing is not considered to be practical. These valves effectively never see service and are subject to conditions (clean, low temperature water) that would minimize their sticking closed. Both valves are stainless steel.~~

~~BS-V30A was opened and inspected for the first time in 1984. It was opened again during the 7R (1988) Outage. It is scheduled to be opened during the 11R Outage (9/95). BS-V30B was disassembled during the 6R (1986) and 9R (1991) Outages. All inspections have shown the valves to be in excellent condition.~~

~~Given the valves' materials of construction, lack of service, and the results of previous disassembly inspections, this relief is justified. Similar relief was approved by the NRC for the previous IST interval in a letter dated December 27, 1989. The NRC granted relief based on GPU Nuclear letters dated June 7, 1988 and April 17, 1989. Inspection results from disassembly subsequent to our previous correspondence provide even greater justification for reducing the frequency of inspection.~~

NRC Comments on Relief Request V4 and GPU Nuclear Response:

NRC Comment:
Do the valves listed in these relief requests (V3, V4, and V5) meet the guidance for extending valve inspection frequency due to extreme hardship as described in GL 89-04, Position 2 and NUREG-1482, pages A-7 through A-15?

GPU Nuclear Response:
Yes. An evaluation has been prepared in accordance with NUREG 1482, "Guidelines for Inservice Testing at Nuclear Power Plants," NRC Staff Position 2, Page A-8, which states that disassemble/inspection can be longer that once every six years, if the following information is developed:

1. Disassemble and inspect each valve in the valve grouping and document in detail the condition of each valve and the valve's capability to be full-stroked.

2. A review of industry experience, for example, as documented in NPRDS regarding the same type of valve in similar service.

3. A review of the installation of each valve addressing the "EPRI Applications Guidelines for Check Valves in Nuclear Power Plants" for problematic locations.

The evaluation,[4] which is documented in the IST program, allows one valve (BS-V30A or BS-30B) to be disassembled/inspected every other refueling outage, alternating between A and B valves. Therefore, relief is not needed.

---

[4] GPU Nuclear Memorandum 3310-96-0007, dated March 13, 1996.

RELIEF REQUEST NO. V 5 (WITHDRAWN)

~~Tag No.~~	~~Component~~	~~Category~~	~~Type~~	~~Actuator~~
~~CO V175A~~	~~EFW Pump Brg Cooling Return Chk Vlv~~	~~C~~	~~Lift Chk~~	~~NA~~
~~CO V175B~~	~~EFW Pump Brg Cooling Return Check Valve~~	~~C~~	~~Lift Chk~~	~~NA~~

~~Code Section from Which Relief is Requested~~
~~Specific relief is requested from the full flow test requirement of NRC Generic Letter 89 04, Position 1. While the system is tested quarterly, it is not possible to determine that both valves open.~~

~~Alternate Test Description~~
~~These valves form a group as described in Generic Letter 89 04, Position 2 and NUREG 1482 in that they are identical, mounted the same manner, and see the same service. One valve in the group will be disassembled and inspected and the disc manually exercised every other refueling outage alternating between A and B, such that each valve is disassembled every fourth outage. This results in a disassembly frequency of approximately every 8 years for a specific valve. If the inspected valve is found to be unacceptable, the other valve in the group will be inspected during the same outage.~~

~~Basis for Relief Request~~
~~These stop check valves direct flow from the EFW Pump seal supply back to the EFW Pump Suction header. The return flow from all three EFW Pumps is combined and then flow is directed through these parallel valves. The closed function is required to prevent draining one Condensate Storage Tank (CST) through the other CST if a failure of one tank is postulated. Full flow testing would require that two of three EFW Pumps (one motor driven and the turbine driven) operate simultaneously with one Condensate Storage Tank/EFW Suction Header isolated (to the opposite motor driven EFW Pump). This test is not practical because isolation of a header and Condensate storage tank with two pumps operating is undesirable.~~

~~Both valves were disassembled and inspected during Refueling Outages 8R (January 1990) and 9R (October 1991); both were found to be in good condition. CO V175A was again disassembled and inspected during 10R (September 1993) and found in good condition. CO V175B is scheduled to be disassembled and inspected in 11R (September 1995). These valves operate in clean water and are of a simple design. Disassembly and inspection every other outage is sufficient to demonstrate the operational readiness of these valves.~~

NRC Comments on Relief Request V5 and GPU Nuclear Response:

NRC Comment:
Do the valves listed in these relief requests (V3, V4, and V5) meet the guidance for extending valve inspection frequency due to extreme hardship as described in GL 89-04, Position 2 and NUREG-1482, pages A-7 through A-15?

GPU Nuclear Response:
Yes. An evaluation has been prepared in accordance with NUREG 1482, "Guidelines for Inservice Testing at Nuclear Power Plants," NRC Staff Position 2, Page A-8, which states that disassemble/inspection can be longer that once every six years, if the following information is developed:

1. Disassemble and inspect each valve in the valve grouping and document in detail the condition of each valve and the valve's capability to be full-stroked.

2. A review of industry experience, for example, as documented in NPRDS regarding the same type of valve in similar service.

3. A review of the installation of each valve addressing the "EPRI Applications Guidelines for Check Valves in Nuclear Power Plants" for problematic locations.

The evaluation,[5] which is documented in the IST program, allows one valve (CO-V175A or CO-V175B) to be disassembled/inspected every other refueling outage, alternating between A and B valves. Therefore, relief is not needed.

---

[5] GPU Nuclear Memorandum 3310-96-0007, dated March 13, 1996.

RELIEF REQUEST NO. VG 1 (UNCHANGED)

Code Section from Which Relief is Requested
Generic relief is requested from OM-10, §§4.2.2.3(a) and 4.1 regarding test frequency.

Alternate Test Description
These tests will be performed each refueling outage instead of the Code specified frequency "at least once every two years."

Basis for Relief Request
The refuel cycle for TMI-1 is nominally two years. Several of the valves requiring leak testing cannot be tested with the plant operating. If, due to an intermediate outage(s), the refueling cycle exceeds two years, the code requirement could require a shutdown simply to test the certain valves. This is impractical. Testing each refueling is a reasonable alternative.

Typically, valve position verification is done more frequently than once every two years. Some valves must be stroked to verify position. Of these, several cannot be stroked with the plant operating. As **described** above, the refuel cycle may extend beyond two years. Position verification using the code specified frequency, could cause the plant to be shut down. Position verification at least every refueling is a reasonable alternative to at least every two years.

Refueling interval testing has been extended to accommodate the 24 month refueling interval. This change was made by a license amendment for all safety related equipment required to be tested each refueling interval. In some cases the regulations have been changed to accommodate 24 month refueling intervals without the need for an exemptions However, for IST position verification tests, the ASME Code has not been changed and therefore written specific relief is required. The fact that this does not represent a significant change is sufficient to justify this relief.

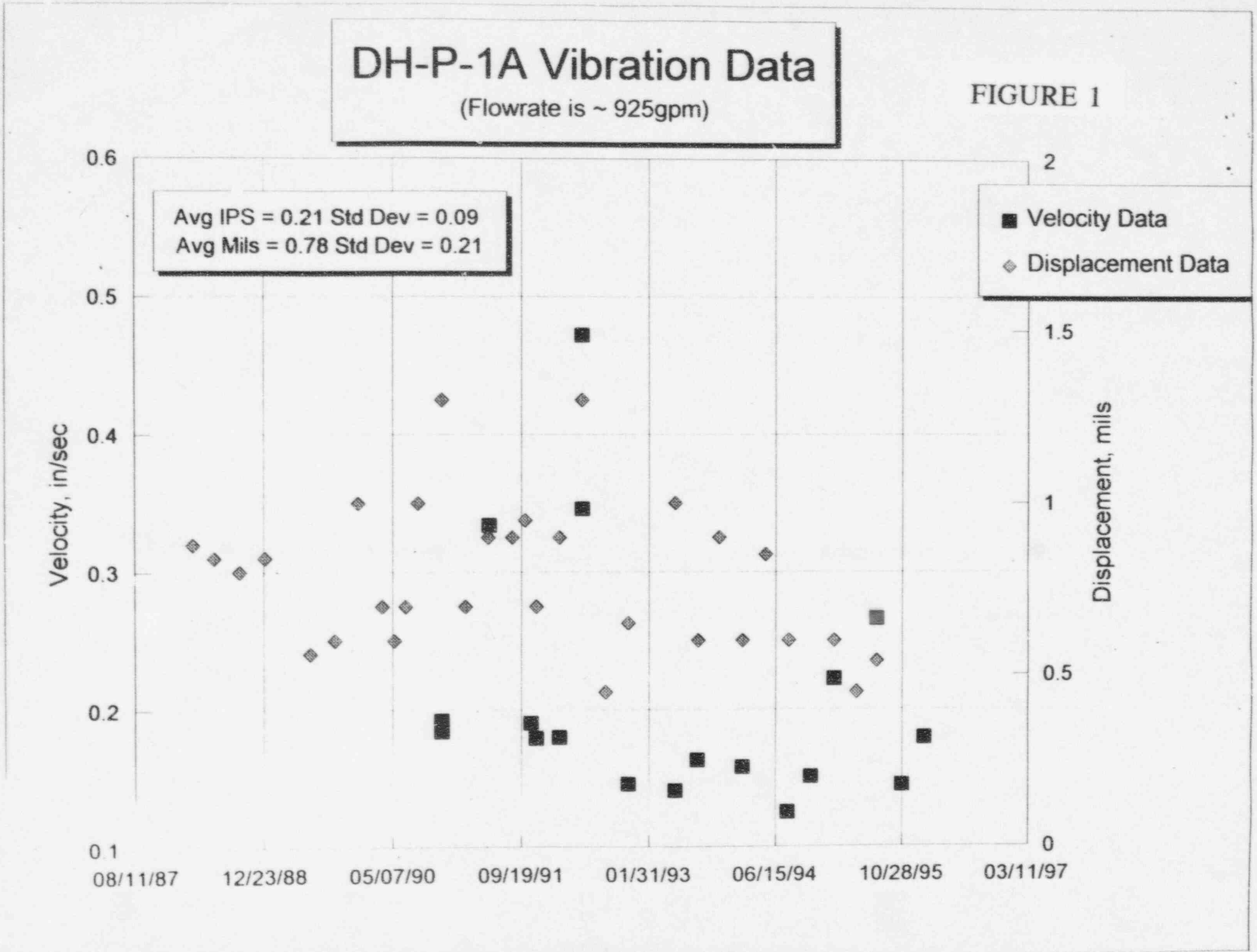
NRC Comments on Relief Request VG1 and GPU Nuclear Response:

NRC Comment:
Licensee should list valves that are within the scope of this relief request.

GPU Nuclear Response:
The valves included within the scope of this relief request are those IST valves listed in Table B-1 where a Position Verification Test is required at the 2-year frequency and are either inaccessible during plant operation or where position verification is only appropriate during the refueling interval test. Although the list may change with changes to the IST program in accordance with 10 CFR 50.59, the current list of these valves is as follows:

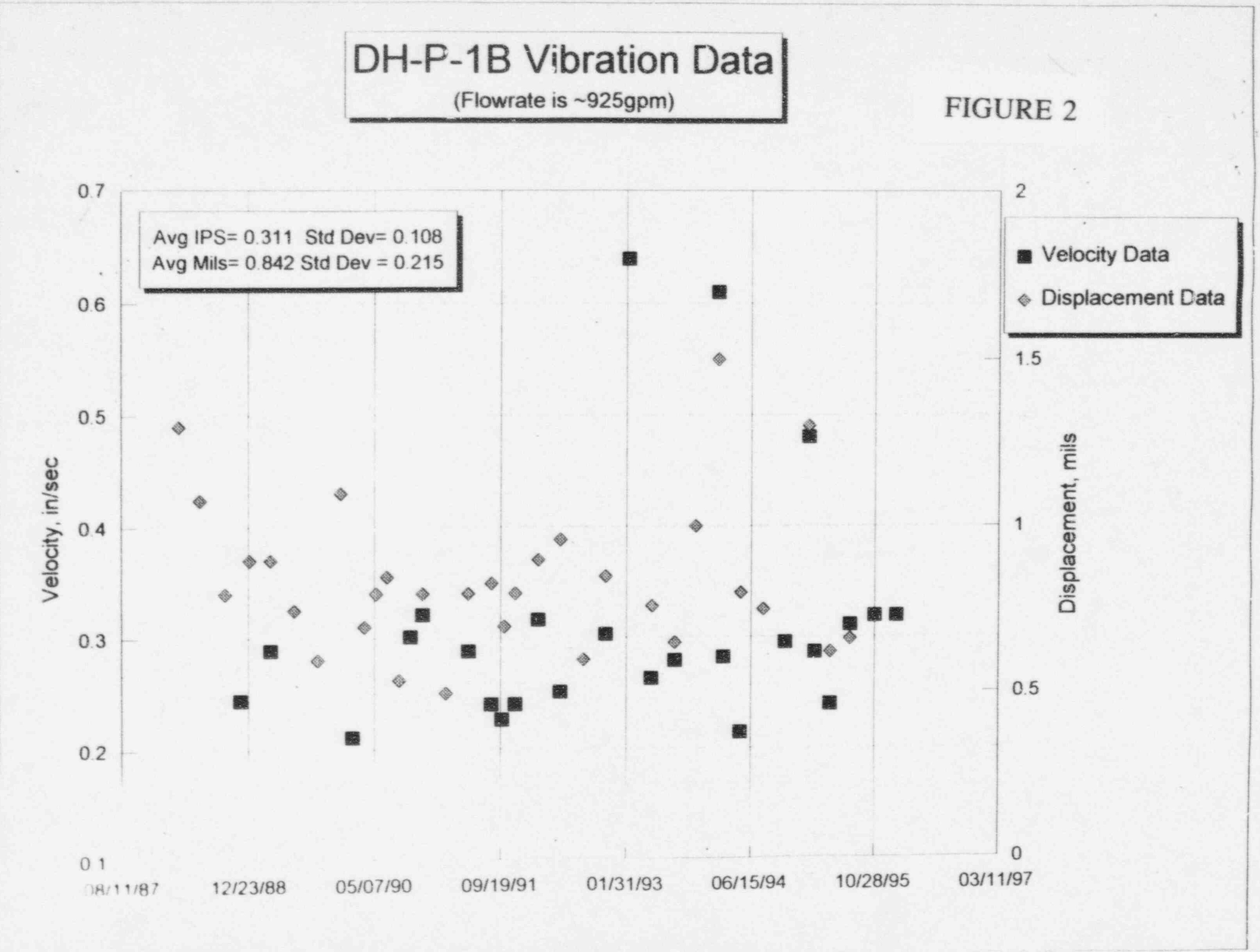
AH-V1A/B/C/D	CF-V20A/B	DH-V4A/B	HM-V4A/B	MS-V1A/E/C/D
CA-V4A/B	CM-V1	DH-V5A/B	HR-V22A/B	MS-V2A/B
CA-V5A/B	CM-V2	DH-V6A/B	HR-V23A/B	MS-V8A/B
CA-V189	CM-V3	EF-V2A/B	IC-V2	MU-V2A/B
CF-V1A/B	CM-V4	HM-V1A/B	IC-V3	MU-V3
CF-V2A/B	DH-V1	HM-V2A/B	IC-V4	MU-V10
CF-V19A/B	DH-V2	HM-V3A/B	IC-V6	MU-V12

MU-V14A/B	NR-V1A/B/C	RB-V2A	RC-V43	WDL-V49
MU-V16A/B/C/D	NR-V2	RB-V7	RC-V44	WDL-V50
MU-V18	NR-V4A/B	RC-RV2	RR-V1A/B	WDL-V61
MU-V20	NR-V6	RC-V2	RR-V3A/B/C	WDL-V62
MU-V25	NS-V4	RC-V4	RR-V4A/B/C/D	WDL-V303
MU-V26	NS-V15	RC-V28	RR-V5	WDL-V304
MU-V36	NS-V35	RC-V40A/B	RR-V10A/B	WDL-V534
MU-V37	NS-V52A/B/C	RC-V41A/B	WDG-V3	WDL-V535
MU-V51	NS-V53A/B/C	RC-V42	WDG-V4	



DH-P-1A Vibration Data
(Flowrate is ~ 925gpm)
Avg IPS = 0.21 Std Dev = 0.09
Avg Mils = 0.78 Std Dev = 0.21
Velocity Data
Displacement Data
Velocity, in/sec
Displacement, mils
0.6
0.5
0.4
0.3
0.2
0.1
2
1.5
1
0.5
0
08/11/87
12/23/88
05/07/90
09/19/91
01/31/93
06/15/94
10/28/95
03/11/97

FIGURE 1



DH-P-1B Vibration Data
(Flowrate is ~925gpm)
Avg IPS= 0.311 Std Dev= 0.108
Avg Mils= 0.842 Std Dev = 0.215
Velocity Data
Displacement Data
Velocity, in/sec
Displacement, mils
0.7
0.6
0.5
0.4
0.3
0.2
0.1
2
1.5
1
0.5
0
08/11/87
12/23/88
05/07/90
09/19/91
01/31/93
06/15/94
10/28/95
03/11/97

FIGURE 2