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10 CFR 50.55a

March 26, 2004
5928-04-20098

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

Three Mile Island, Unit 1
Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Submittal of a Request for Relief to the Requirements of 10CFR 50.55a Concerning Fourth Ten-Year Interval Inservice Testing Program

Attached for your review and approval are proposed relief requests in accordance with 10CFR 50.55a, associated with the fourth ten-year interval Inservice Testing (IST) program for Three Mile Island (TMI), Unit 1. Based on a start date of September 23, 2004, the TMI, Unit 1 fourth interval IST program complies with the requirements of the ASME OM Code-1998, Code for Operation and Maintenance of Nuclear Power Plants, including the OMb Code-2000 Addenda. The fourth interval will conclude on September 22, 2014.

We request your review and approval by September 15, 2004.

If you have any questions or require additional information, please do not hesitate to contact us.

Very truly yours,

D. S. Kelker /FOR

Michael P. Gallagher
Director, Licensing and Regulatory Affairs
AmerGen Energy Company, LLC

Attachments: 1) Pump Relief Request Index
2) Pump Relief Requests
3) Valve Relief Request Index
4) Valve Relief Requests

cc: H. J. Miller, Administrator, Region I, USNRC
D. M. Kern, USNRC Senior Resident Inspector, TMI-1
D. M. Skay, USNRC Senior Project Manager
File No. 02078

A047

ATTACHMENT 1

PUMP RELIEF REQUEST INDEX

(Page 1 of 1)

Relief Request No.	Description
PR-01	Nuclear Service River Water Flow Rate Measurement During Group A Tests
PR-02	Nuclear Service Closed Cooling Water Flow Rate Measurement During Group A Tests
PR-03	Turbine-Driven Emergency Feedwater Pump Comprehensive Pump Testing Requirements
PR-04	Categorization of Decay Heat Removal Pumps as Group B (Power Operation)

ATTACHMENT 2

PUMP RELIEF REQUESTS

10 CFR 50.55a Request Number PR-01

Relief
In accordance with 10 CFR 50.55a(f)(5)(iii) and 10CFR50.55a(a)(3)(i)

1. ASME Code Component(s) Affected

NR-P1A	Nuclear Service River Water Pump A
NR-P1B	Nuclear Service River Water Pump B
NR-P1C	Nuclear Service River Water Pump C

2. Applicable Code Edition and Addenda

ASME OM Code 1998 Edition through 2000 Addenda

3. Applicable Code Requirement(s)

ISTB-5221(b) Group A Test Procedure – The resistance of the system shall be varied until the flow rate equals the reference point.....Alternatively, the flow rate shall be varied until the differential pressure equals the reference point.

ISTB-3400 Frequency of Inservice Tests – An inservice test shall be run on each pump as specified in Table ISTB-3400-1. Table ISTB-3400-1 requires a comprehensive pump test to be performed biennially for Group A pumps.

4. Reason for Request

Pursuant to 10 CFR 50.55a(f)(5)(iii), relief is requested from the requirements of ASME OM Code ISTB-5221(b) and ISTB Table ISTB-3400-1. Due to system design and plant operating requirements, individual pump flow rate cannot be measured during the Group A test as required by ISTB-5221(b).

The flow instrumentation for this system is located in the common discharge header for all three of the subject pumps. The piping configuration does not contain, nor would the system design permit the installation of accurate individual pump flow measuring devices due to the turbulence caused by the valving, strainer and elbow configuration on the discharge of the pumps. TMI has investigated individual annubar instrumentation for this configuration; however, the accuracy and repeatability of using individual annubar instrumentation has not produced results to meet IST instrument requirements.

Since the refueling cycle for Three Mile Island is nominally two years, a situation may exist where the plant may be required to shut down in order to perform the biennial Comprehensive Pump Test on the subject pumps. In the event of an extended intermediate outage, the biennial frequency (once every two years) may be exceeded. Therefore, an alternative is also requested in accordance with 10CFR50.55a(a)(3)(i) to perform the comprehensive pump test on a refueling outage frequency, which will avoid the potential for an unnecessary plant shutdown while testing to a biennial frequency.

The TMI Technical Specifications define Refueling Outage Interval as 24 months. Additionally, Technical Specification 1.25 provides an interval extension of 25% for cycle lengths, which exceed the 24 months. Therefore, the proposed alternative will be similar to the TMI Technical Specifications.

5. Proposed Alternative and Basis for Use

The Nuclear Services River Water Pumps are tested each quarter (see Attachment 1 diagram). The quarterly surveillance is currently performed with two pumps in operation. During the quarterly test, the differential pressure is throttled to the reference value for each pump combination. Vibration data on the motor is recorded and compared to the reference values. Vibration is recorded on the motor bearing in the Horizontal, Vertical and Axial directions. Any deviation from the reference value is compared to the Code acceptance criteria. As expected with vertical line shaft pumps, the vibration measurements recorded have been relatively low and consistent. Therefore, this testing method provides an acceptable level in quality and safety for determining pump performance.

The Nuclear Services River Water Pumps are also tested each refueling outage. During this test, the system is throttled to obtain the reference flow rate (6000 gpm) with a single pump in operation. The pump differential pressure and vibration levels are measured and compared to their reference values. Any deviations from the reference values are compared to the Code acceptance criteria. Therefore, the testing method described above provides an acceptable level in quality and safety for determining pump performance.

As a proposed alternative:

a. TMI will continue to perform quarterly pump testing using a modified Group A test procedure. With two pumps in service, the required group A test parameters will be measured except for flow rate. During this test the differential pressure will be throttled to the reference value. Vibration measurements will then be recorded and compared to their reference values. Deviations from the reference value will be compared with the ranges specified in Table ISTB-5200-1 for Group A tests. Corrective actions will be taken in accordance with ISTB-6200.

b. Each of the subject pumps will be tested individually in accordance with ISTB-5223, Comprehensive Test Procedure during refueling outages.

c. During testing of the subject pumps (quarterly and refueling), TMI will perform full spectrum vibration analysis, which is done above the Code required vibration testing.

d. The comprehensive pump test will be performed on a refueling outage frequency, which will avoid the potential for an unnecessary plant shutdown while testing to a biennial frequency.

Using the provisions of this relief request as an alternative to the specific requirements of ISTB 5221 and Table 3400-1 identified above will provide an acceptable level of quality and safety.

6. Duration of Proposed Alternative

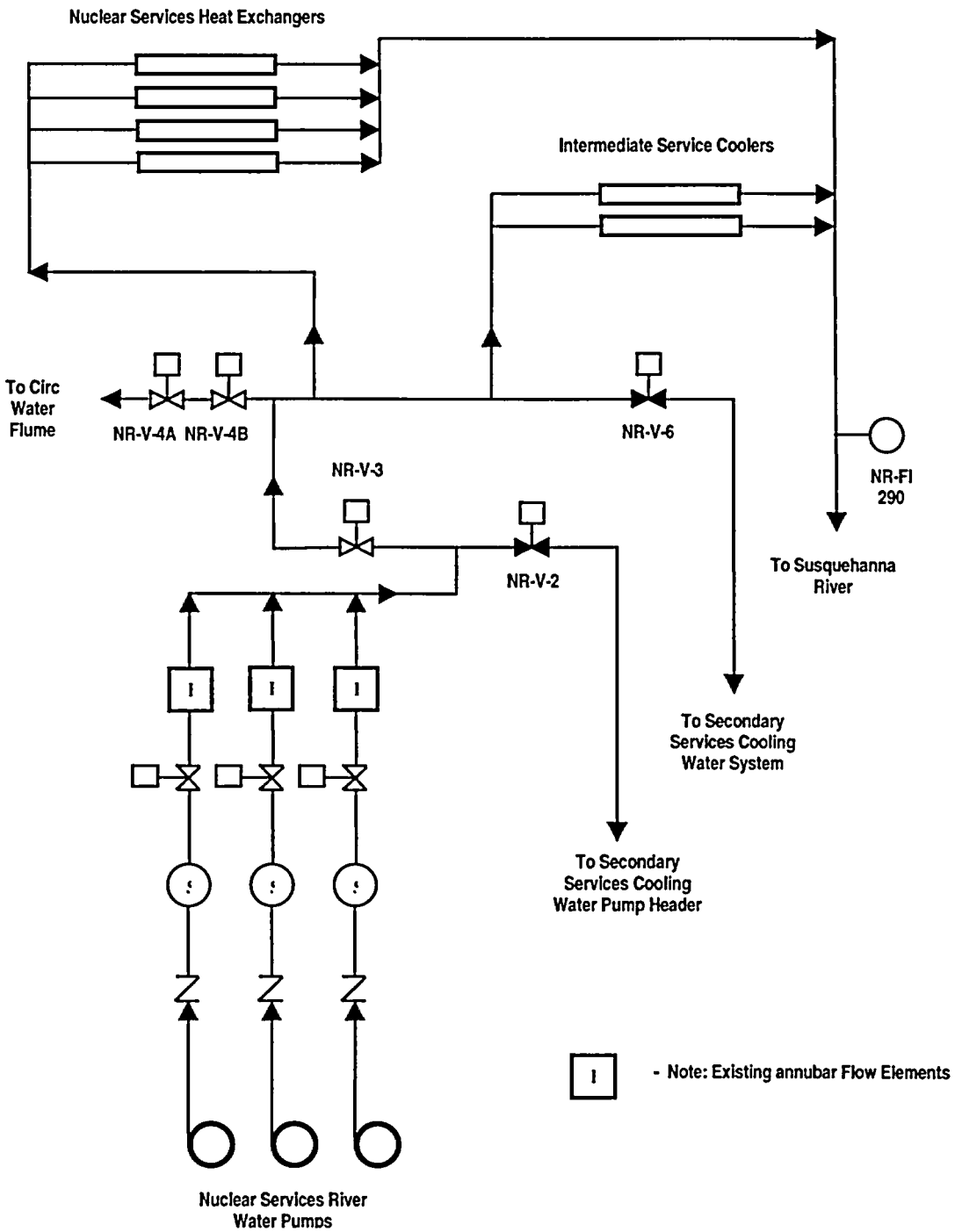
This proposed alternative will be utilized for the 4th 120 month interval.

7. Precedents

This relief request was previously approved for a similar request in third ten-year interval at Three Mile Island as P 3.

10 CFR 50.55a Request Number PR-01

Attachment 1 - Nuclear Services River Water System Diagram


Nuclear Services Heat Exchangers
Intermediate Service Coolers
To Circ Water Flume
NR-V-4A
NR-V-4B
NR-V-6
NR-V-3
NR-V-2
NR-FI 290
To Susquehanna River
To Secondary Services Cooling Water System
To Secondary Services Cooling Water Pump Header
I
- Note: Existing annubar Flow Elements
Nuclear Services River Water Pumps

10 CFR 50.55a Request Number PR-02

Relief
In accordance with 10 CFR 50.55a(f)(5)(iii) and 10CFR50.55a(a)(3)(i)

1. ASME Code Component(s) Affected

NS-P-1A	Nuclear Services Closed Cooling Water Pump 1A
NS-P-1B	Nuclear Services Closed Cooling Water Pump 1B
NS-P-1C	Nuclear Services Closed Cooling Water Pump 1C

2. Applicable Code Edition and Addenda

ASME OM Code 1998 Edition through 2000 Addenda

3. Applicable Code Requirement

ISTB-5121(b) - "Group A Test Procedure" – The resistance of the system shall be varied until the flow rate equals the reference point.....Alternatively, the flow rate shall be varied until the differential pressure equals the reference point.

ISTB-3400 - "Frequency of Inservice Tests" – An inservice test shall be run on each pump as specified in Table ISTB-3400-1. Table ISTB-3400-1 requires a comprehensive pump test to be performed biennially for Group A pumps.

4. Reason for Request

Pursuant to 10 CFR 50.55a(f)(5)(iii), relief is requested from the requirement of ASME OM Code ISTB-5121(b) and ISTB Table ISTB-3400-1.

Due to system design and plant operating requirements, it is not practical to reduce the number of pumps in service to one to allow for single-pump testing during power operation as required by ISTB-5121(b). Also, individual pump flow rates cannot be measured during the Group A test. The flow instrumentation for this system is located in the common discharge header for all three of the subject pumps. The piping configuration does not contain, nor would the system design permit the installation of accurate individual pump flow measuring devices due to the turbulence caused by the valving and elbow configuration on the discharge of the pumps.

There were no provisions originally designed in the system to measure individual pump flowrate. Individual suction and discharge pressure gauges are installed at each pump, allowing for measurement of differential pressure for inservice testing. A flow instrument is installed in the common discharge header.

Since the refueling cycle for Three Mile Island is nominally two years, a situation may exist where the plant may be required to shut down in order to perform the biennial Comprehensive Pump Test on the subject pumps. In the event of an extended intermediate outage, the biennial frequency (once every two years)

may be exceeded. Therefore, an alternative is requested in accordance with 10CFR50.55a(a)(3)(i) to perform the comprehensive pump test on a refueling outage frequency, which will avoid the potential for an unnecessary plant shutdown while testing to a biennial frequency.

The TMI Technical Specifications define Refueling Outage Interval as 24 months. Additionally, Technical Specification 1.25 provides an interval extension of 25% for cycle lengths, which exceed the 24 months. Therefore, the proposed alternative will be similar to the TMI Technical Specifications.

5. Proposed Alternative and Basis for Use

Individual pump flow cannot be measured during normal quarterly operations since individual flow instrumentation does not exist. Also, two pumps are normally required to be inservice to provide adequate cooling for system components. The current testing methodology of testing paired-combinations of pumps near two-pump design flowrate provides an adequate basis for identifying and evaluating degraded pump performance.

To comply with the ISTB requirement for measuring individual pump flow rates on a quarterly basis, a modification of the system would be required.

The Nuclear Services Closed Cooling Water pumps are currently tested each quarter. The quarterly surveillance is currently performed with paired-pumps in operation (A-B, A-C, B-C). During the quarterly test, the differential pressure is throttled to the reference value for each pump combination. Total system flowrate is recorded and compared to the reference values. Also, vibration data on the pump is recorded and compared to the reference values. Vibration is recorded on the pump bearing in the Horizontal, Vertical and Axial directions. Any deviation from the reference value is compared to the Code acceptance criteria. As expected with centrifugal pumps, the vibration measurements recorded have been relatively low and consistent. Therefore, this testing method provides an acceptable level in quality and safety for determining pump performance.

The Nuclear Services Closed Cooling Water Pumps are also tested individually each refueling outage. The pump differential pressure and vibration levels are measured and compared to their reference values. Any deviations from the reference values are compared to the Code acceptance criteria. Therefore, this testing method provides an acceptable level in quality and safety for determining pump performance.

As proposed alternatives:

a. TMI will continue to perform quarterly pump testing using a modified Group A test procedure. With two paired-pumps in service, the required group A test parameters will be measured except for individual pump flow rate. During this test the differential pressure will be throttled to the reference value.

Vibration measurements will then be recorded and compared to their reference values. Deviations from the reference value will be compared with the ranges specified in Table ISTB-5100-1 for Group A tests. Corrective actions will be taken in accordance with ISTB-6200.

b. Each of the subject pumps will be tested individually in accordance with ISTB-5123, Comprehensive Test Procedure during refueling outages.

c. During testing of the subject pumps (quarterly and refueling), TMI will perform full spectrum vibration analysis, which is above Code required vibration testing.

d. The comprehensive pump test will be performed on a refueling outage frequency, which will avoid the potential for an unnecessary plant shutdown while testing to a biennial frequency.

Using the provisions of this relief request as an alternative to the specific requirements of ISTB-5121 and Table ISTB-3400-1 will provide an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This proposed alternative will be utilized for the 4th 120 month interval.

7. Precedents

A similar relief request was approved for TMI in the second ten-year interval at Three Mile Island as 2.6, as documented in the U. S. Nuclear Regulatory Commissions Safety Evaluation Report dated October 3, 1986.

A similar relief was also submitted for the third interval in correspondence to the U. S. Nuclear Regulatory Commission dated September 24, 2003 and March 10, 2004 for the third interval.

10 CFR 50.55a Request Number PR-03

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected

   EF-P-1 Turbine Driven Emergency Feedwater Pump

2. Applicable Code Edition and Addenda

   ASME OM Code 1998 Edition through 2000 Addenda

3. Applicable Code Requirement

   ISTB-3300(e)(1) – Reference values shall be established within +/- 20% of pump design flow rate for the comprehensive test.

4. Reason for Request

   Pursuant to 10 CFR 50.55a, "Codes and standards", paragraph (a)(3)(i), relief is requested from the requirement of ASME OM Code ISTB-3300(e)(1). The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety. Specifically, relief is requested from ISTB-3300(e)(1) in meeting the specified +/- 20% of design flow (736 gpm [920 gpm x 80%]) during the comprehensive pump testing. The specified +/- 20% of pump design flow rate can not be achieved for the subject pump during normal quarterly Group B testing or during Comprehensive testing without introducing large volumes of highly oxygenated water into the once-through-steam generators which increases the potential for corrosion of the steam generators. The design flow rate of the turbine driven Emergency Feedwater Pump is 920 gpm at 2750 feet of developed head (see Attachment 1).

   This relief is intended to provide an alternative to supplying a large volume of highly oxygenated water into the Once-Through-Steam Generators (OTSGs), which increases the potential for corrosion of the steam generators.

5. Proposed Alternative and Basis for Use

   Historically, the pump testing methodology has adequately measured pump performance, as demonstrated below. The current refuel testing procedure tests flow through one (of two) fully-open control valves to the OTSG. Figures 1 and 2 provide a simplified diagram of the EFW system flowpath and steam supply to EF-P-1. During refueling interval testing of EF-P-1, the pump is tested individually by injecting water from the condensate storage tanks into a OTSG. The OTSG is depressurized during this test. Flow to the steam generator is throttled until total flow delivered to the OTSG is at least the minimum accident-required flow rate of

290 GPM at a minimum pump head of 1165 PSIG. These testing requirements were submitted in response to a request for additional information related to TMI, Unit 1 technical specification change request number 279 (Core Protection Safety Limit, 20% tube plugging), in a letter dated May 21, 1999, and found to be acceptable.

During this test, flow is also set at approximately 500 gpm and the differential pressure is measured. Differential pressure data at 500 gpm from the last three refueling outage tests is presented in Attachment 2, EF-P-1 Test Data. This data has been speed corrected to 3800 rpm using pump laws for variable speed pumps with constant impeller diameter and plotted against the original manufacturers curve. Additionally, recent quarterly test data points are also presented on this attachment. It can be seen that the pump is operating at or near the original manufacturers curve for both the quarterly and refueling outage tests. Additionally, vibration data collected during the inservice tests has never exceeded 2.5 times the reference values or the 0.325 in/sec absolute value specified by ISTB.

Preoperational startup test data from 1974 is plotted against the original manufacturers curve in Attachment 3. Additionally, during TMI's 12R refueling outage, an EF-P-1 flow capacity test was performed. The data from 12R is also plotted against the original manufacturers curve in Attachment 3. This data closely matches the preoperational data and the manufacturers curve. As expected for a standby pump, the EF-P-1 pump has not degraded.

This relief will minimize the volume of fluid introduced from the condensate storage tanks into the Once-Through Steam Generators (OTSGs), thus minimizing the potential for corrosion of the OTSGs (specifically the tubes). Specifically, administrative controls are in place to minimize the amount of dissolved oxygen in the OTSGs. The inventory used during the testing of EF-P-1 originates in condensate storage tanks. These tanks are vented to the atmosphere and normally have a dissolved oxygen concentration between 6,000 ppb and 7,000 ppb.

During lay-up, dissolved oxygen concentrations in the OTSGs are normally maintained less than 100 ppb to minimize the potential for corrosion of the steam generator. Secondary chemistry controls require the steam generator volume to have a dissolved oxygen concentration below this level. To minimize the impact of this highly oxygenated water injected into the OTSGs (under non-emergency conditions), a number of steps are taken.

To reduce the dissolved oxygen concentration in the condensate storage tanks, a nitrogen tanker truck is specially staged in order to sparge dissolved oxygen from the tank water. Large volumes of nitrogen are injected just prior to the testing of all three emergency feedwater pumps. This normally reduces the dissolved oxygen concentration to between 600 ppb to 1,200 ppb. In addition, the chemistry of the OTSGs is adjusted to maximize the concentration of oxygen-scavenging chemicals. Also, the levels of the steam generators are maintained as high as possible during the emergency feedwater testing in order to minimize the dilution of these chemicals. Finally, in the event that dissolved oxygen limits

are exceeded during the testing, administrative controls require that the steam generator be drained and refilled under a nitrogen blanket and the water chemistry is adjusted as necessary to maintain the water chemistry within allowable limits within 48 hours. These actions are in agreement with TMI, Unit 1's commitments to the NRC for protecting steam generator tube integrity in accordance with NEI 97-06 and associated industry guidelines.

By minimizing the volume of fluid introduced from the condensate storage tanks into the OTSGs, the potential for corrosion of the OTSGs (specifically the tubes) is minimized.

As an alternative to testing at +/- 20 % of design flow, TMI will test EF-P-1 at a reference flow rate of approximately 500 gpm versus 736 gpm (80% of design flow of 920) each refueling outage. All other requirements of the comprehensive test will be followed. At this reference point of 500 gpm, the characteristic curve for the pump is essentially the same slope as at the Code required 736 gpm. Pump degradation as noted by measuring differential pressure can be detected for a given flow rate reference value.

The reference flow rate of 500 gpm corresponds to 54.4 % of pump design flow. At the reference conditions the flow values are currently at a point on the curve that is effective for monitoring and detecting degradation. Testing at this reference point has resulted in very repeatable measurements. Any degradation in pump performance at the set flow rate can be recognized or detected by a substantial change in measured pump differential pressure.

To establish the flow rate within +/- 20 % of design would require a flow rate of at least 736 gpm. Establishing the flow at 736 gpm does not increase the ability to detect degradation or assess pump conditions since the slope of the pump curve is essentially a straight line constant from 300 gpm to 800 gpm as shown below. Therefore, testing at higher flows does not increase the ability to detect hydraulic degradation.

$$\text{Slope (400 to 600 gpm)} = \frac{(2950 - 2850)\ \text{ft}}{(400 - 600)\ \text{gpm}} = \frac{-0.5\ \text{feet of head}}{\text{gpm}}$$

$$\text{Slope (600 to 800 gpm)} = \frac{(2850 - 2750)\ \text{ft}}{(600 - 800)\ \text{gpm}} = \frac{-0.5\ \text{feet of head}}{\text{gpm}}$$

To compensate for testing the EF turbine driven pump at a reduced flow rate during the comprehensive test, as required by ISTB-3300(e)(1), additional activities will be performed as follows to assess operational readiness and determine pump health.

Full spectrum bearing vibration analysis as well as oil sampling and analysis is performed as part of the preventative maintenance program. Finally, during each shift the operations staff inspects these pumps to ensure that no problems are present. Based on the full spectrum bearing vibration analysis, the oil sampling and analysis, operational inspections, continued quarterly Group B testing and comprehensive testing within 54% of design pump flow during refueling outages, an accurate assessment of pump health and operational readiness is assured.

As a proposed alternative:

a. TMI will test EF-P-1 at a reference flow rate of approximately 500 gpm each refueling outage. All other requirements of the comprehensive test will be followed.

b. During comprehensive testing of the subject pumps, TMI will perform spectrum vibration analysis, which is above Code required vibration testing.

The proposed alternative testing coupled with OTSG water chemistry concerns provides for: a) testing verification of pump performance and identification of degradation, b) verification of piping flowpath capability to deliver accident design flow rates and c) appropriate secondary chemistry precautions to protect OTSG tube integrity.

6. Duration of Proposed Alternative

This proposed alternative will be utilized for the 4th 120 month interval.

7. Precedents

- Similar relief request P-6 was previously approved for North Anna Power Station on January 28, 2002. Docket Nos. 50-338 and 50-339 (TAC Nos. MB2221and MB2222).
- Similar relief request PR-1 was previously approved for Seabrook Station on May 30, 2003. Docket No. 50-443 (TAC No. MB6676).

FIGURE 1 - EMERGENCY FEEDWATER

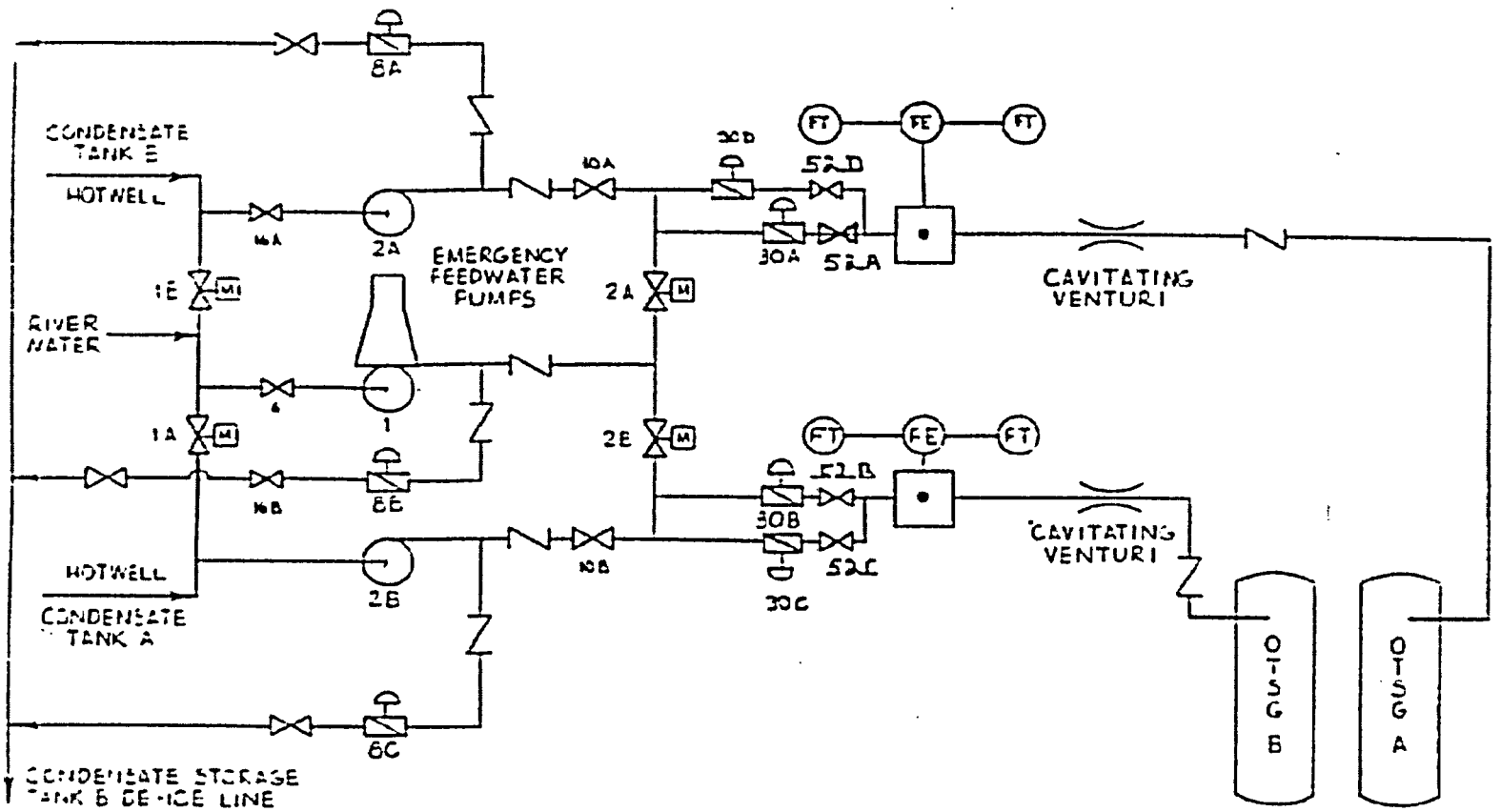
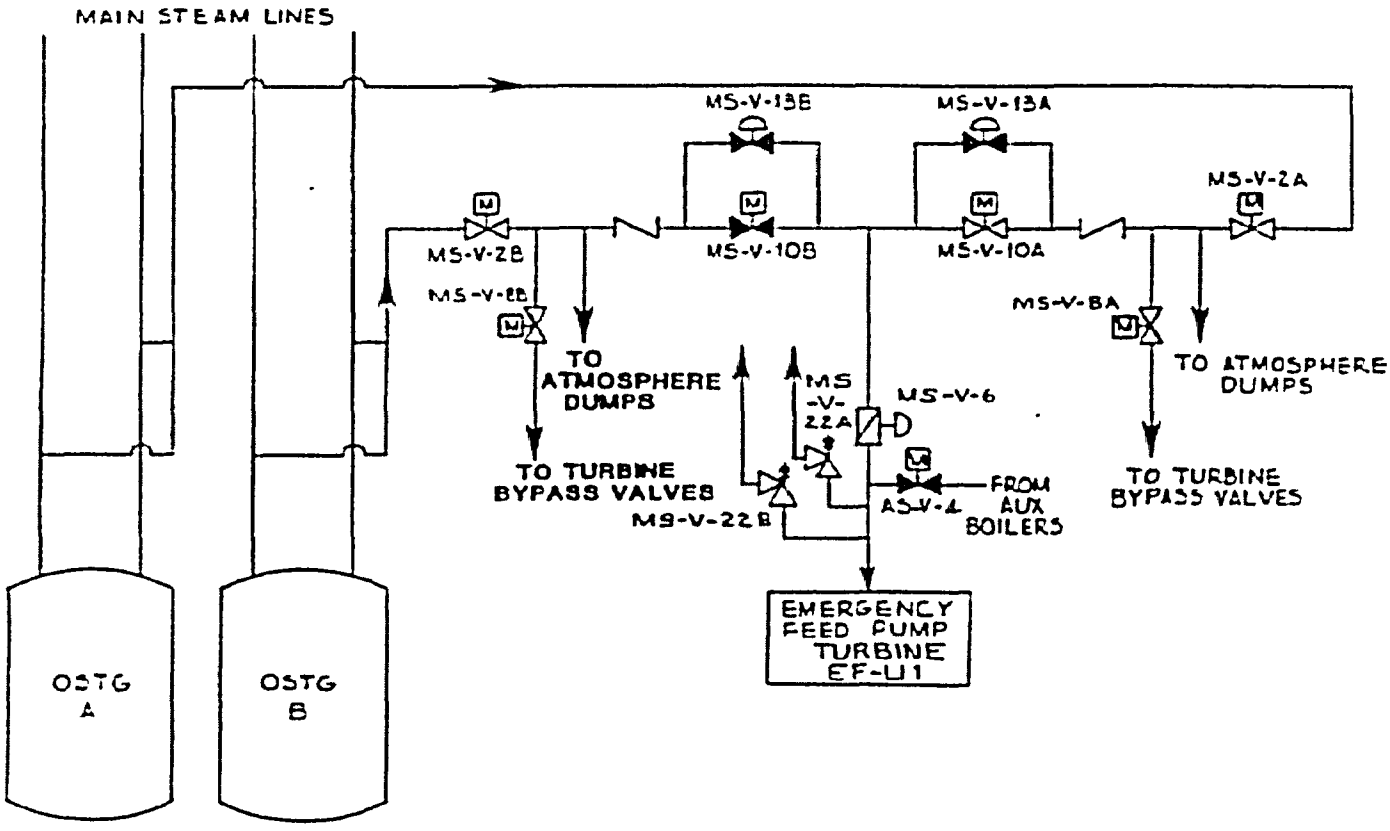
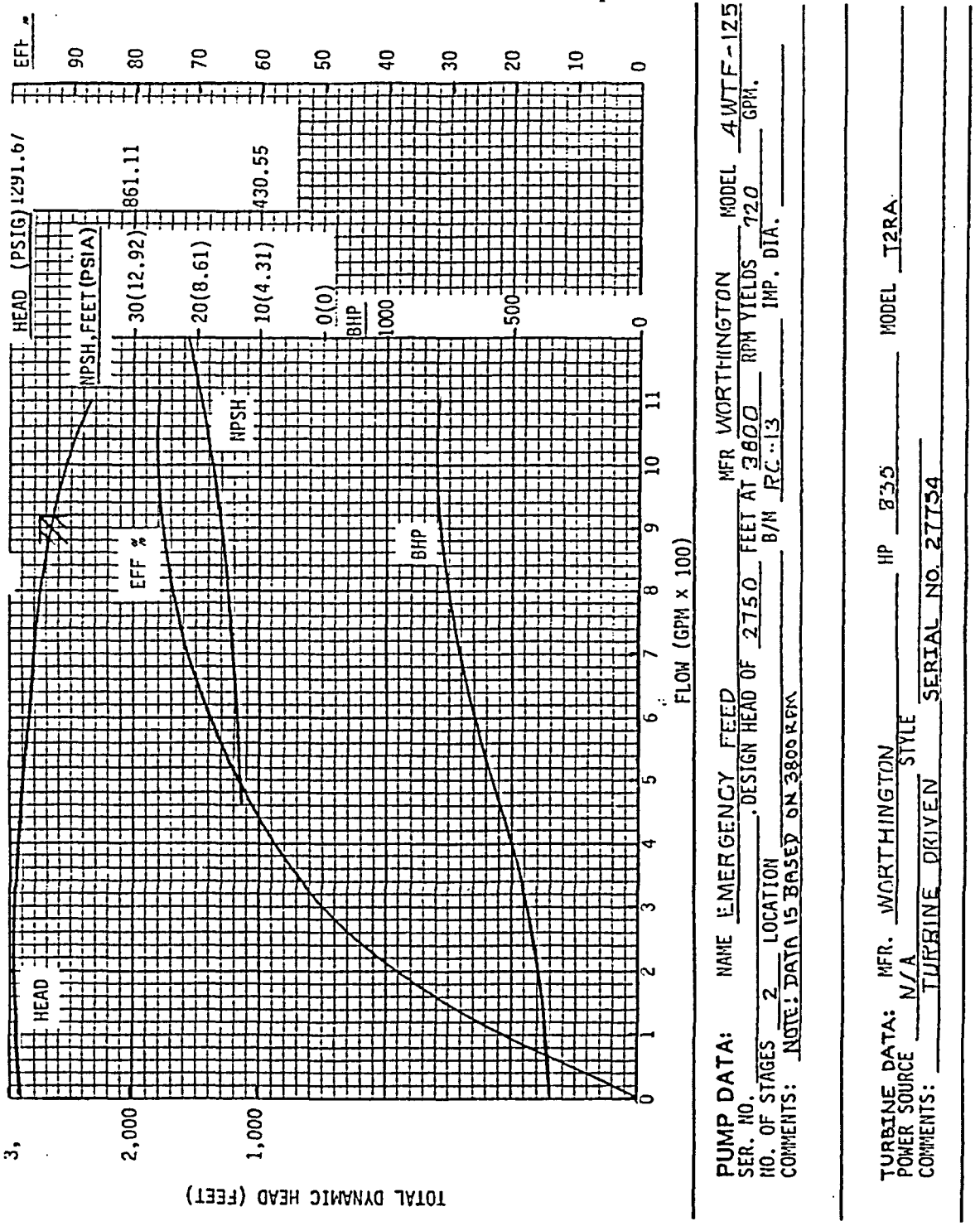


FIGURE 2 - STEAM FLOW TO EF-P-1



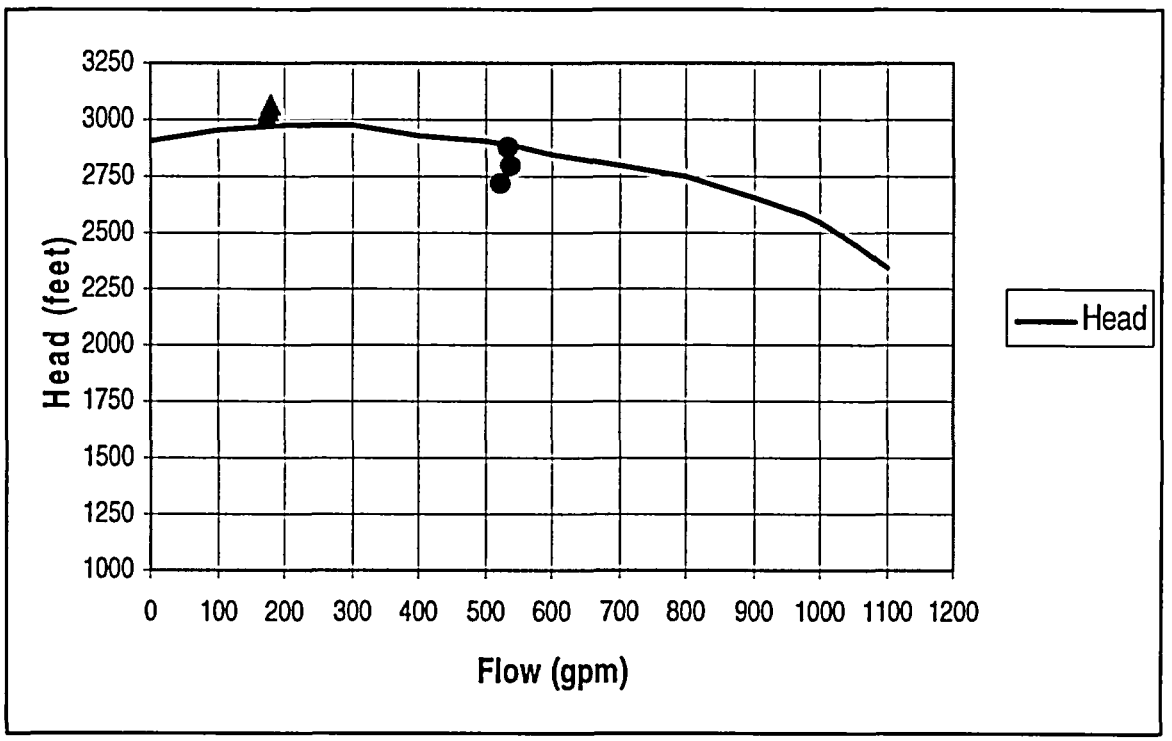
Attachment 1

EF-P-1 Manufacturers Pump Curve



Attachment 2

EF-P-1 Test Data.

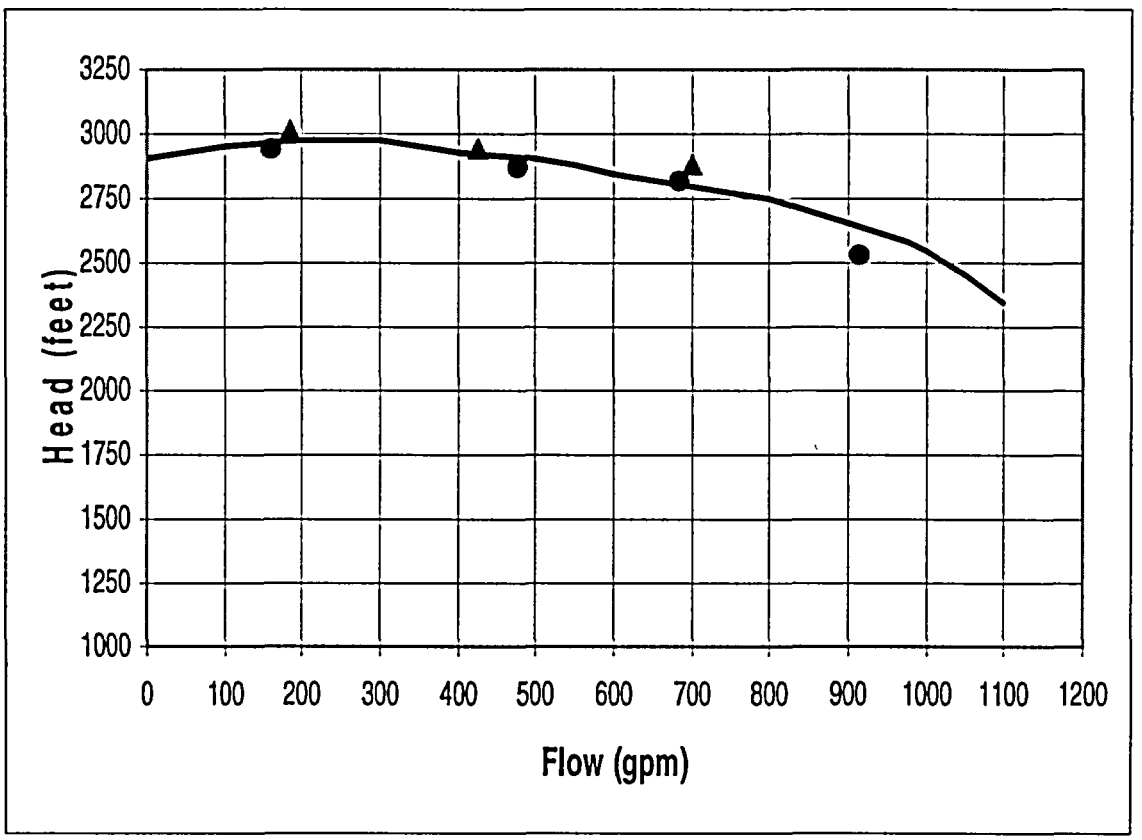


▲ **Quarterly Test** ● **Refueling Outage Test**

Quarterly Test Data			**Refueling Test Data**		
Date	Flow	TDH	Date	Flow	TDH
5/07/03	182	3003	10/14/99	536	2787
11/18/03	180	3027	11/28/01	533	2894
2/03/04	180	3025	12/01/03	526	2729

Attachment 3

EF-P-1 Preoperational Startup Test Data and 12R Outage Data



▲ 12R Refueling Data

● 1974 Preoperational Data

10 CFR 50.55a Request Number PR-04

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected

   DH-P-1A Decay Heat Removal Pump A
   DH-P-1B Decay Heat Removal Pump B

2. Applicable Code Edition and Addenda

   ASME OM Code 1998 Edition through 2000 Addenda

3. Applicable Code Requirement

   ISTB-1400(b), "identify each pump to be tested in accordance with the rules of this Subsection and categorize it as either a group A or group B pump and list the pumps in the plant records (see ISTB-9000). A pump that meets both group A and group B definitions shall be categorized as a group A pump."

4. Reason for Request

   Pursuant to 10 CFR 50.55a, "Codes and Standards", paragraph (a)(3)(ii), relief is requested from the requirement of ASME OM Code ISTB-1400(b). This relief will result in testing the Decay Heat Removal pumps as group B during power operations versus group A. This proposed relief will result in a lower potential for pump degradation due to pump wear while still being capable of measuring pump performance. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

   The Decay Heat Removal pumps meet the categorization requirements of group A pumps in that they are operated routinely during plant shutdowns and refueling outages. However, these pumps also meet the requirements of group B, in that during normal operation (reactor critical) they are not operated except for testing.

   During normal power operations, the Decay Heat Removal pump is in a standby condition and is considered an essential part of the Emergency Core Cooling System (ECCS). The pump starts automatically upon receipt of a safety injection signal taking suction from the Borated Water Storage Tank during the injection phase of an accident. The pump is then aligned to take suction from the containment sump during the recirculation phase of an accident. The pump discharges to the reactor coolant system via the decay heat removal heat coolers.

ASME ISTB-1400(b) states that if a pump meets both group A and group B definitions, it shall be categorized as a group A pump. The Decay Heat Removal pumps are currently tested during normal operation, using the minimum flow recirculation loop. This test is similar to a group B test in that the pump is operated at low flow conditions (approximately 850 gpm).

The design flow rate of the Decay Heat Removal Pumps is 3000 gpm. This flow rate can only be achieved during shutdown periods when injection into the reactor coolant system at a reduced pressure is possible.

Classifying these pumps as group B during power operation minimizes the time required to perform quarterly testing. The 1998/2000 ASME Code testing requirements eliminate the two-minute minimum pump run-time for quarterly group B pump tests. Eliminating the minimum pump run-time requirement and the requirement to record differential pressure and vibration levels is expected to slightly reduce the length of each quarterly pump test and the accompanying unavailability time for these pumps. Since these pumps are not operated routinely during plant operation, except for required surveillance testing, there is no time- or wear-related degradation mechanism that would warrant performing more detailed quarterly tests on DH-P-1A/B.

NUREG/CP-0137, Vol. 1, Proceedings of the Third NRC/American Society of Mechanical Engineers (ASME) Symposium on Valve and Pump Testing, includes a paper entitled, "Description of Comprehensive Pump Test Change to ASME Code, Subsection ISTB." This paper details the philosophy of classifying pumps as group A or group B. According to the author, the intent of having different test requirements for different pump groups is to relate the requirements for the amount and degree of quarterly performance monitoring to the amount of degradation expected based on pump operation.

Testing the decay heat removal pumps quarterly as group A pumps during power operation is contrary to the philosophy of the referenced paper. Quarterly group A testing subjects these pumps to increased test requirements and performance monitoring. Also, this testing introduces the potential for more degradation due to pump wear (caused by low-flow operation) at the time when they are standby pumps and would not otherwise be subject to operation-induced degradation. Group A testing during power operation may be more detrimental to the long-term health of these components than Group B testing.

5. Proposed Alternative and Basis for Use

TMI proposes that the decay heat removal pumps (DH-P-1A and DH-P-1B) be tested as standby pumps (group B) during power operation and as continuously operating pumps (group A) during refueling operations. During refueling operations, the comprehensive pump test may be substituted for a quarterly group A test that comes due. TMI further proposes that any time a comprehensive pump test is performed, the Code-required quarterly low-flow test (group B) requirement may be deleted for that quarter.

Using the provisions of this relief request as an alternative to the specific requirements of ISTB-1400(b) identified above will provide adequate indication of pump performance and continue to provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i) we request relief from the specific ISTB requirements identified in this request.

This alternative provides an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This proposed alternative will be utilized for the 4th 120 month interval.

7. Precedents

Similar relief request PR-12 was previously approved for Calvert Cliffs Nuclear Power Plant on May 16, 2002 (TAC Nos. MB3782 and MB3783).

ATTACHMENT 3

VALVE RELIEF REQUEST INDEX

(Page 1 of 1)

Relief Request No.	Description
VR-01	2-Year Frequency of Seat Leakage Testing of Valves which are not Containment Isolation Valves
VR-02	2-Year Frequency of Remote Position Indication Verification Tests for Specific Valves

ATTACHMENT 4

VALVE RELIEF REQUESTS

10 CFR 50.55a Request Number VR-01

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected

Valve Number	Class	Category	Function
MU-V-14A/B	2	A/C	MU Pump Suction from BWST Stop Check Valve
MU-V-112	2	A/C	Makeup Tank Outlet Check Valve
CF-V-4A/B	1	A/C	Core Flood Tank Outlet Check Valve
CF-V-5A/B	1	A/C	Core Flood Tank and Decay Heat Pump Discharge Check Valve
NR-V-2	3	A	NR to SR Header Isolation Valve
NR-V-4A/B	3	A	Deicing Makeup Valve
NR-V-6	3	A	NR & SR Cross Connection Isolation Valve

2. Applicable Code Edition and Addenda

ASME OM Code 1998 Edition through 2000 Addenda

3. Applicable Code Requirement

ISTC-3630(a) – "Leakage Rate for Other Than Containment Isolation Valves", regarding Frequency. Tests shall be conducted at least once every 2 years.

4. Reason for Request

Pursuant to 10 CFR 50.55a, "Codes and Standards", paragraph (a)(3)(i), relief is requested from the requirement of ASME OM Code ISTC-3630(a). ISTC-3630(a) requires a leak test to be performed at least once every two (2) years. TMI proposes to test these valves on a refueling outage frequency in order to avoid an unnecessary shutdown. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

The subject valves are all categorized as A or A/C since they have a seat leakage requirement to fulfill their specific function in the closed direction. These valves are not containment isolation valves.

Leakage testing on the subject valves is performed during refueling outages when the associated systems can be removed from service.

Since the refueling cycle for Three Mile Island is nominally two years, a situation may exist where the plant may be required to shut down in order to perform the leakage test using the two (2) year frequency defined in the code. Therefore, an alternative is requested in accordance with 10CFR50.55a(a)(3)(i) to perform the leakage test on a refueling outage frequency, which will avoid the potential for an unnecessary plant shutdown while testing to a two (2) year frequency.

The TMI Technical Specifications define Refueling Outage Interval as 24 months. Additionally, Technical Specification 1.25 provides an interval extension of 25% for cycle lengths, which exceed the 24 months. Therefore, the proposed alternative will be similar to the TMI Technical Specifications.

5. Proposed Alternative and Basis for Use

The seat leakage test for the subject valves will be performed each refueling outage.

Performance of the seat leakage tests at a frequency of every refueling outage versus once every two years provides reasonable assurance of the seat leakage requirements of the subject valves.

Using the provisions of this relief request as an alternative to the specific requirements of ISTC-3630(a) identified above will provide adequate indication of valve performance and continue to provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i) we request relief from the specific ISTC requirements identified in this request.

This alternative provides an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This proposed alternative will be utilized for the 4th 120 month interval.

7. Precedents

This relief request was previously approved for the 3rd Ten Year Interval at Three Mile Island as VG1.

10 CFR 50.55a Request Number VR-02

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected

Valve Number	Class	Category	Function
AH-V-1Athru D	2	A	RB PURGE OUTLET ISOL VALVES
CA-V-4A/B	2	A	OTSG FW SAMPLE VALVES
CA-V-5A/B	2	A	OTSG FW SAMPLE VALVES
CA-V-189	2	A	RECLAIMED WATER TO RB VLV
CF-V-1A/B	2	B	CORE FLOOD TANK DISCHARGE ISO VALVES
CF-V-2A/B	2	A	CORE FLOOD TANK SAMPLE ISO VLVS
CF-V-19A/B	2	A	MU TO CORE FLOOD TANK ISO VLVS
CF-V-20A/B	2	A	CORE FLOOD TANK SAMPLE ISOL VLVS
CM-V-1	2	A	RB ATMOSPHERE SAMPLE VALVE
CM-V-2	2	A	RB ATMOS SAMPLE RETURN VLV
CM-V-3	2	A	RB ATMOSPHERE SAMPLE VALVE
CM-V-4	2	A	RB ATMOS SAMPLE ISOL VLV
DH-V-1	1	B	DH SUCTION ISOLATION FROM "B" HOT LEG
DH-V-2	1	B	DH DROP LINE/PUMP SUCTION
DH-V-4A/B	2	B	DH-PUMP DISCHARGE ISOL VLVS
DH-V-5A/B	2	B	DECAY HEAT SUCTION VALVES FROM BWST
DH-V-6A/B	2	B	RB SUMP RECIRC SUCTION VLVS
EF-V-2A/B	3	B	EFW PUMP HEADER CROSS CONNECT VALVES
HM-V-1A/B	2	A	H2 MONITOR OUTLET ISOL
HM-V-2A/B	2	A	H2 MONITOR INLET ISOL
HM-V-3A/B	2	A	H2 MONITOR OUTLET ISOL
HM-V-4A/B	2	A	H2 MONITOR INLET ISOL
HR-V-22A/B	2	A	RB EXHAUST TO H2 RECOMB
HR-V-23A/B	2	A	H2 RECOMB RETURN ISOL VLVS
IC-V-2	3	A	ICCW COOLANT RETURN VALVE
IC-V-3	2	A	ICCW COOLANT RETURN VALVE
IC-V-4	2	A	IC ISOL COOLANT SUPPLY
IC-V-6	2	A	IC COOLANT SUPPLY TO CRDM
MS-V-1A thru D	2	B/C	OTSG MS ISOL VALVES
MS-V-2A/B	2	B	OTSG TO EF PUMP & TURBINE BYPASS VALVES
MU-V-2A/B	1	A	LETDOWN CLR OUTLET VLVS
MU-V-3	2	A	LETDOWN COOLER ISOL VALVE
MU-V-14A/B	2	A/C	MU PUMP SUCTION FROM BWST STOP CHECK
MU-V-16A thru D	2	B	HPI CONTROL VALVES
MU-V-18	2	A	CHARGE LINE ISOL VALVE
MU-V-20	2	A	RCP SEAL WATER ISOL VLV

Valve Number	Class	Category	Function
MU-V-25	2	A	RCP SEAL RETURN ISOL VALVE
MU-V-26	2	A	RCP SEAL RETURN LETDOWN ISOL
MU-V-36	2	B	MU PUMPS RECIRC ISOLATION VALVE
MU-V-37	3	B	MU PUMPS RECIRC ISOL VALVE
NR-V-1A/B/C	3	B	NUC RIVER PUMP DISCH VALVES
NR-V-2	3	A	NR TO SR HEADER ISOLATION VALVE
NR-V-4A/B	3	A	DEICING MAKEUP VALVES
NR-V-6	3	A	HX VAULT CROSS CONNECT BETWEEN NR & SR
NS-V-4	2	A	RCP MOTOR COOLER RETURN VALVE
NS-V-15	2	A	RCP MOTOR COOLER RETURN VALVE
NS-V-35	3	A	RCP MOTOR COOLER RETURN VALVE
NS-V-52A/B/C	2	B	AH-MOTOR COOLER SUPPLY
NS-V-53A/B/C	2	B	AH-MOTOR COOLER RETURN
RB-V-2A	2	A	RB NORMAL AIR COOL SUP VLV
RB-V-7	2	A	RB NORMAL COOL RETURN ISOL
RC-V-2	1	B	PORV (RC-RV2) ISOLATION VALVE
RC-RV-2	1	C	PZR PILOT OPERATED RELIEF VALVE (PORV)
RC-V-4	1	B	DECAY HEAT PRESS SPRAY LINE ISOL VALVE
RC-V-28	1	B	PZR VENT TO RCDT ISOLATION VALVE
RC-V-40A/B	1	B	HOT LEG HIGH POINT VENT TO RCDT AND ATM
RC-V-41A/B	1	B	HOT LEG VENT TO RCDT AND ATMOSPHERE
RC-V-42	1	B	REACTOR VESSEL VENT TO RB ATMOSPHERE
RC-V-43	1	B	REACTOR VESSEL VENT TO RB ATMOSPHERE
RC-V-44	1	B	PRESSURIZER HIGH POINT VENT ISO VALVE
RR-V-1A/B	3	B	RR-PUMP DISCHARGE VALVES
RR-V-3A/B/C	2	B	RBEC COIL INLET VALVES
RR-V-4A thru D	2	B	RBEC COIL OUTLET VLVS
RR-V-5	3	B	RB COOLING COIL DISCHARGE BYPASS VALVE
RR-V-10A/B	3	B	RR PUMP RECIRC MINFLOW BYPASS VALVES
WDG-V-3	2	A	RB VENT HEADER VALVE
WDG-V-4	2	A	RB VENT HEADER VALVE
WDL-V-303	2	A	RC DRAIN PUMP DISCH VALVE
WDL-V-304	2	A	RC DRAIN PUMP DISCH VALVE
WDL-V-534	2	A	RB SUMP DRAIN TO AUX BLDG
WDL-V-535	2	A	RB SUMP DRAIN TO AUX BLDG

2. Applicable Code Edition and Addenda

ASME OM Code 1998 Edition through 2000 Addenda

3. Applicable Code Requirement

ISTC-3700, "Position Verification Testing". Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated.

4. Reason for Request

Pursuant to 10 CFR 50.55a, "Codes and Standards", paragraph (a)(3)(i), relief is requested from the requirement of ASME OM Code ISTC-3700. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

The subject valves are power-operated valves with remote position indication. To perform the remote position verification, the valve must be cycled from its normal position (open or closed) to the full stroke position. Local observation of the valve position is verified with the remote position indication at least once every 2 years. The valve is then cycled to its normal position. Local observation of the valve is then verified again to match the remote position indication.

The subject valves listed are in systems or portions of a system, which typically cannot be isolated during normal power operations or cold shutdowns. Additionally, many of these valves are inside the reactor building and would require a containment entry to verify local valve position.

Since the refueling cycle for Three Mile Island is nominally two years, a situation may exist where the plant may be required to shut down in order to perform the 2 year position indication verification testing on these subject valves. In the event of an extended intermediate outage, the two year frequency may be exceeded. Therefore, an alternative is requested in accordance with 10CFR50.55a(a)(3)(i) to perform the verification on a refueling outage frequency, which will avoid the potential for an unnecessary plant shutdown while testing to a two (2) year frequency. The TMI Technical Specifications define Refueling Outage Interval as 24 months. Additionally, Technical Specification 1.25 provides an interval extension of 25% for cycle lengths, which exceed the 24 months. Therefore, the proposed alternative will be similar to the TMI Technical Specifications.

5. Proposed Alternative and Basis for Use

The position indication verification test as described above for the subject valves will be performed each refueling outage.

Using the provisions of this relief request as an alternative to the specific requirements of ISTC-3700 identified above will provide adequate indication of valve performance and continue to provide an acceptable level of quality and safety, and will avoid the potential for an unnecessary plant shutdown while testing to a two (2) year frequency. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i) we request relief from the specific ISTC requirements identified in this request.

This alternative provides an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This proposed alternative will be utilized for the 4th 120 month interval.

7. Precedents

This relief request was previously approved for the 3rd Ten Year Interval at Three Mile Island as VG1.