UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO REQUEST FOR RELIEF VRR-4B TO

THE INSERVICE TESTING PROGRAM

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

The *Code of Federal Regulations*, 10 CFR 50.55a, requires that inservice testing (IST) of certain American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (the Code) and applicable addenda, except where relief has been requested and granted or proposed alternatives have been authorized by the Commission pursuant to 50.55a(a)(3)(i), (a)(3)(ii), or (f)(6)(i). In order to obtain authorization or relief, the licensee must demonstrate that (1) conformance is impractical for its facility, (2) the proposed alternative provides an acceptable level of quality and safety, or (3) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Section 50.55a authorizes the Commission to approve alternatives and to grant relief from ASME Code requirements upon making the necessary findings. NRC guidance contained in Generic Letter (GL) 89-04, Guidance on Developing Acceptable Inservice Testing Programs, identified acceptable alternatives to the Code requirements, as set forth in Positions 1, 2, 6, 7, 9, and 10 provided the licensee follows the guidance delineated in the applicable position. Additional guidance for establishing acceptable IST programs is given in GL 89-04 and NUREG-1482.

By letter dated January 2, 1998, the Wisconsin Electric Power Company (the licensee) submitted request for relief VRR-4B from a commitment made in a previously approved relief request (VRR-4) for the Point Beach Nuclear Plant, Units 1 and 2, IST program. The IST program was based on the criteria of 1986 Edition of Section XI of the ASME Code, for the third 10-year interval that began December 31, 1990, for both units.

Relief request VRR-4B is evaluated below. VRR-4 was evaluated in NRC's April 17, 1992, and October 28, 1993, safety evaluations, and approved per GL 89-04, Position 2. The approval included the condition that, if the licensee develops nonintrusive techniques for exercising these valves in accordance with the requirements of Section XI, the use of disassembly and inspection in lieu of testing should be discontinued, except as required for preventive maintenance or valve internal inspection. VRR-4A, which requested an extension to the inspection interval for Unit 1 safety injection (SI) check valve SI-867A from the spring 1996 until the 1997 Unit 1 refueling outage was approved in a staff SE dated February 22, 1996, as an alternative to the current schedule.

2.0 RELIEF REQUEST VRR-4B

Relief is requested from the ASME Code quarterly exercising requirements of IWV-3522 for the Unit 1 SI check valve SI-867A. This request extends the previous one-time extension from the 1996 refueling outage until the Unit 1 1998 refueling outage. Compliance with the previous relief request for an extension until the Unit 1 1997 refueling outage was not possible since Unit 1 did not have a refueling outage during 1997. Relief Request VRR-4B proposes to extend a 6-year disassembly and inspection interval for SI-867A on Unit 1 by 2 years from the spring 1996 refueling outage. After 1998, the interval would revert to 6 years.

3.0 LICENSEE'S BASIS FOR REQUEST

The SI check valve opens to provide a flow path from the SI pumps and the SI accumulator to the reactor coolant system (RCS) cold leg under accident conditions. It closes to isolate the SI system from the RCS, protects the SI system from potential damage caused by overpressurization, and is tested in accordance with Technical Specification (TS) 15.3.1. The valve is not installed in a problematic location based on its orientation in the piping system.

Currently (per relief requests VRR-4 and -4A), valves SI-842A and SI-867A are each required to be disassembled, inspected, and manually stroked once every 6 years, in rotating sequence. Valve SI-867A was scheduled for maintenance during the spring 1996 outage. The NRC SE dated February 22, 1996, relating to relief Request VRR-4A granted an extension to the 6-year interval until the 1997 refueling outage. The basis of the relief was that no other activities except disassembly and inspection of the check valve required mid-loop operation of the RCS and the history of no degradation of valve operability or performance in any disassembly and inspection or full-flow test performed on any similar valves (three additional valves in Unit 1 and four valves in Unit 2). Because of an extended shutdown during 1997, the plant was not refueled and no operations were performed that reduced RCS level to mid-loop operation, inspection of SI-867A did not occur.

This disassembly, inspection, and manual stroke testing of Unit 1 SI-867A is required to be performed during a refueling outage when RCS level can be lowered below the cold leg while fuel still remains in the reactor core. The following information was provided by the licensee per NRC GL 89-04, Position 2, in support of this relief request. A total of seven other similar valves have been disassembled or full-flow tested within the last 4 years. Six of the seven valves have been disassembled within the last 6 years. To date, no degradation of valve operability or performance has been noted in any disassembly and inspection or full-flow test performed on these valves. The following table lists each specific valve, the individual work order (WO) or outage activity (OA) under which the work was performed. and the completion date:

	Unit 1	
SI-00842A	WO 45881	4/14/93
	WO 890172	4/11/90
	WO 872759	4/14/88
SI-00842B	WO 45639	4/14/93
	WO 890174	4/21/90
SI-00867A	WO 3637	5/1/90
	WO 890176	4/24/90
	WO 8[illegible]2[illegible]55	4/15/88
SI-00867B	WO 9700761	Spring 97*
	OA 8739	Spring 96*
	OA 8739	Spring 95*
	OA 8739	Spring 94*
	WO 890178	4/21/90
	Unit 2	
SI-00842A	WO 9510056	10/17/95
	WO 890173	10/5/89
	WO 872760	10/16/87
SI-00342B	WO 9510057	10/17/95
	WO 890175	11/4/89
SI-00867A	WO 9510060	10/21/95
	WO 890177	10/5/89
	WO 872753	10/20/87
SI-00867B	WO 9610739	Fall 96*
	OA 8739	Fall 95*
	OA 8739	Fall 94*
	WO 50730	10/8/93
	WO 890179	11/3/89

*Full flow test

An industry-wide search, performed January 2, 1998, utilizing the Nuclear Plant Reliability Data System (NPRDS - a component maintenance/failure database managed by the Institute of Nuclear Power Operations) on similar valves also indicated no failures, although leakage through the seat was reported in 34 instances, including 3 instances at Point Beach. Allowable leakage values are given in TS Table 15.3.16-1.

As required in VRR-4 and -4A, this valve has been successfully partially stoked open and shut at each refueling outage and at each cold shutdown in which an Event V test was required. In addition, this valve has successfully passed its seat leakage test in accordance with Point Beach TS 15.3.16, "Reactor Coolant System Pressure Isolation Valve Leakage Tests."

4.0 PROPOSED ALTERNATIVE TESTING

Valve SI-0867A will be disassembled, inspected, and manually stroked after an 8-year interval.

5.0 EVALUATION

The category A/C valve SI-867A opens to provide a flow path from the SI pumps and SI accumulators to the RCS cold leg during certain accidents. The valve is normally closed. In the closed position, the valve functions as an RCS pressure isolation valve.

This request is for relief from impractical Code requirements as previously approved for Relief Requests VRR-4 and -4A in NRC safety evaluations dated April 17, 1992, October 28, 1993, and February 22, 1996. This relief request proposes to extend the 6-year disassembly and inspection interval for Unit 1 SI-867A by 2 years from the spring 1996 refueling outage. This valve is scheduled to be disassembled and inspected during the spring 1998 refueling outage. There have been no intervening refueling outages since the spring 1996 outage. After spring 1998, the interval would revert to 6 years.

The relief request indicates that the disassembly and inspection program for this valve is in accordance with GL 89-04, Position 2. The related Position 2 guidance states:

> Extension of the valve disassembly and inspection interval to one valve every other refueling outage or expansion of the group size above four valves should only be considered in cases of extreme hardship where the extension is supported by actual in-plant data from previous testing. In order to support extension of the valve disassembly/inspection intervals to longer than once every 6 years, licensees should develop the following information:
>
> a. 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.
>
> b. Review industry experience, for example, as documented in NPRDS, regarding the same type of valve used in similar service.
>
> c. Review the installation of each valve addressing the "EPRI [Electric Power Research Institute] Applications Guidelines for Check Valves in Nuclear Power Plants" for problematic locations.

Until the end of the Unit 1, 1998 refueling outage, the partial-stroke exercising and the Event V leakage testing provide information on the valve, and the disassembly and inspection of the remaining similar valves provide a measure of monitoring for degrading conditions. The licensee indicates that an NPRDS search on similar valves indicated no failures, although leakage past the seat was reported in 34 instances, including 3 instances at Point Beach. Additionally, this valve is not installed in a "problematic location" based on the orientation of the valve in the piping system.

The valve is partial-stroke exercised each refueling outage and during any cold shutdowns that require an Event V test per plant technical specifications. A leakage test is performed at least every refueling outage. Under the circumstances, not extending the 6-year interval for this valve would require the unit to shut down prior to the next scheduled refueling outage, reduce RCS inventory, and expose the reactor core to potential damage just to meet the current inspection schedule. This is impractical and would constitute an extreme hardship as described in of GL 89-04, Position 2.

5.0 CONCLUSION

The staff has evaluated the information provided by the licensee in its relief request and proposed alternative testing, i.e., disassembly, inspection, and partial stroke testing of Unit 1 check valve SI-867A after an 8-year interval. The licensee requests an extension of the disassembly and inspection interval until the spring 1998 refueling outage. Information from disassembly and inspection of similar check valves provides confidence that the short extension will not be adverse to the public health and safety. The staff has determined that for Relief Request VRR-4B, the requirements of the Code are impractical as stated above, and therefore the relief is granted pursuant to 10 CFR 50.55a(f)(6)(i) until the end of the 1998 Unit 1 refueling outage (no later than June 30, 1998). The relief granted is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest given due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. The licensee should maintain documentation supporting the basis for VRR-4B.

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Date: January 2, 1998