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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSERVICE TESTING PROGRAM RELIEF REQUEST

ILLINOIS POWER COMPANY

CLINTON POWER STATION

DOCKET NO. 50-461

INTRODUCTION

The Code of Federal Regulations, 10 CFR 50.55a, requires that inservice testing (IST) of certain 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 and applicable addenda, except where relief has been requested and granted or proposed alternatives have been authorized by the Commission pursuant to 10 CFR 50.55a (f)(6)(i), (a)(3)(i), or (a)(3)(ii). 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 a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Section 50.55a (f)(4)(iv) provides that inservice tests of pumps and valves may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed, and subject to Commission approval. NRC guidance contained in Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," provided alternatives to the Code requirements determined to be acceptable to the staff and authorized the use of the alternatives in Positions 1, 2, 6, 7, 9, and 10 provided the licensee follow the guidance delineated in the applicable position. When an alternative is proposed which is in accordance with GL 89-04 guidance and is documented in the IST program, no further evaluation is required; however, implementation of the alternative is subject to NRC inspection.

Section 50.55a authorizes the Commission to grant relief from ASME Code requirements or to approve proposed alternatives upon making the necessary findings. The NRC staff's findings with respect to granting or not granting the relief requested or authorizing the proposed alternative as part of the licensee's IST program are contained in this Safety Evaluation (SE).

This relief request references the first 10-year interval for the Clinton Power Station (CPS), stating that the IST program is based on the 1980 edition through Winter 1981 addenda of ASME Section XI. CPS had implemented the requirements of OMa-1988, Part 10, Section 4.2.2.3 for leak rate testing of Category A valves in accordance with the guidance in NUREG-1482. The licensee's relief request was transmitted to the staff in a letter dated June 18, 1998, and supplemented by a letter dated September 4, 1998.

Relief Request

The licensee has requested relief from the requirements of OMa-1988, Part 10, Section 4.2.2.3. This section defines the frequency for leak rate testing of Category A valves. A one-time extension of the 2-year test interval is requested for leak rate testing of the 21 valves listed below:

1E12-F008	1E12-F009	1E12-F023	1E12-F041A
1E12-F041B	1E12-F041C	1E12-F042A	1E12-F042B
1E12-F042C	1E12-F050A	1E12-F050B	1E12-F053A
1E12-F053B	1E21-F005	1E21-F006	1E22-F004
1E22-F005	1E51-F013	1E51-F066	1B21-F001
1B21-F002			

Relief is requested to extend the test interval for each valve until the next refueling outage.

Licensee's Basis for Requesting Relief

The licensee states:

> As a result of the current extended outage, the 2-year limit for some of the valves may be exceeded prior to plant startup. In any case, the limit will be exceeded for all of the valves before the next scheduled refueling outage. Extension of the test interval would preclude the necessity of testing the valves twice during the current outage and instead allow the next required test to be performed during the next refueling outage scheduled for CPS.
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> All of the valves were tested with high pressure (1000 psig) water between October 29, 1996, and January 28, 1997. CPS has been in Mode 4 (cold shutdown) or Mode 5 (refueling) since before this testing was performed. Thus, the valves have not been exposed to operating conditions or the typical degradation processes associated with a cycle of operation. Therefore, little additional assurance of these valves' ability to perform their leakage safety function would be gained by reperformance at this time.
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> Reperformance of these tests at this time places an additional burden on CPS with no compensating increase in the level of quality and safety. Nearly all of these valves are located in high radiation areas, and therefore, leak testing these valves again would result in unnecessary radiation exposure for test personnel. In addition to the time required for test preparation, valve lineups, equipment setup, test performance, and post-test restoration, testing of several of these valves would require significant manipulations of plant equipment, including removal of the drywell head.

Alternative Testing

Some of the 21 pressure isolation valves are also containment isolation valves. As such, these valves will be leak rate tested in accordance with the requirements of 10 CFR 50, Appendix J (Type C test) prior to startup, using air at 9.0 psig as the test medium. The following valves are included:

1E12-F008	1E12-F009	1E12-F023	1E12-F042A
1E12-F042B	1E12-F042C	1E12-F053A	1E12-F053B
1E21-F005	1E22-F004	1E51-F013	

The frequency for performing high pressure water tests of the containment isolation valves will return to once per refueling outage after completion of testing during the next refueling outage, which will occur approximately 18 months after startup from the current plant shutdown.

EVALUATION

Pressure isolation valves (PIVs) are defined as two normally closed valves in series that function to isolate the reactor coolant system from an attached low pressure system. Valves which are defined as containment isolation valves, have additional safety functions for isolation or functioning of a system such as emergency core cooling, or train separation of a safety system to prevent diversion of flow. The Code requires that these Category A valves be seat-leakage tested at a frequency of once every 2 years.

Six of the valves for which relief is requested (1E12-F008, 1E12-F009, 1E12-F050A, 1E12-F053A, 1E12-F050B, 1E12-F053B) are required for establishing the residual heat removal shutdown cooling (SDC) mode. When testing these valves, the normal method for SDC is lost. An alternate method to remove decay heat is the use of the reactor water cleanup system. This system is powered from a non-safety auxiliary power bus.

Currently, the licensee is performing maintenance activities in the switchyard. With the resulting breaker configuration, an electrical anomaly on the transmission line would cause the 4160 volt and 6900 volt non-safety buses to be lost. The loss of non-safety buses would cause a loss of the reactor water cleanup system which, if it was in service as the decay heat removal system, would result in a loss of SDC. Loss of decay heat removal capability represents an unnecessary risk to the plant.

The valves 1E51-F013 and 1E51-F066 are located on the reactor head spray piping. 1B21-F001 and 1B21-F002 are on the reactor vessel head vent line to the drywell. 1E12-F023 is a connection from the B train of residual heat removal to the reactor core isolation cooling system. Testing of these valves would involve removal of the drywell head and disassembly of the reactor vessel head spray and vent piping. The estimated dose that would be received by licensee personnel in performance of these tests is significant.

The remaining valves, 1E12-F041A, 1E12-F042A, 1E12-F041B, 1E12-F042B, 1E12-F041C, 1E12-F042C, 1E21-F005, 1E21-F006, 1E22-F004, and 1E22-F005, are located on the injection lines of the five emergency core cooling systems. These valves serve as the injection line motor-operated shutoff valves and check valves.

All of the valves for which the licensee requests relief have successfully passed their Category A tests which were performed near the beginning of the extended outage. Since that time, they have not been in service at normal operating temperature or pressure. All of the valves, with the exception of 1B21-F001 and 1B21-F002, are constructed from carbon steel with stellite seat rings and discs. 1B21-F001 and 1B21-F002 have hard faced seats and discs made of cobalt based alloy A567. These materials are corrosion resistant. The materials of construction combined with the valves not being exposed to normal degradation processes provides sufficient assurance of their operational readiness.

Results from the past six leak rate tests performed on the valves were provided. All the measured leakage values were well below acceptance criteria of 5 gpm with the exception of the latest test of valve 1E12-F053B. However, 1E12-F053B also serves as a containment isolation valve and therefore will undergo Appendix J testing prior to startup. Minimal leakage during Appendix J testing would provide reasonable assurance of the valve's operational readiness.

Compliance with the Code would result in hardship without a compensating increase in safety. Reasonable assurance exists of the operational readiness of the valves due to their materials of construction, their testing history, and the fact that they have not been exposed to normal degradation processes associated with power operation. Therefore, the licensee is authorized to defer testing until the next refueling outage.

CONCLUSION

On the basis of the above evaluation, the staff concludes that the proposed alternative to the requirements of OMa-1988, Part 10, Section 4.2.2.3 for the 21 Category A reactor coolant system pressure isolation valves is authorized pursuant to 10 CFR 50.55a (a)(3)(ii) based on the determination that compliance with the specified requirements results in a hardship without a compensating increase in the level of quality and safety. The alternative is authorized for an interim period from the date of the SE until the end of the 7th refueling outage.

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