AEOD TECHNICAL REVIEW REPORT*

UNIT: Browns Ferry Nuclear Plant, Unit 1
DOCKET NO.: 50-259
LICENSEE: Tennessee Valley Authority
NSSS/AE: General Electric/Tennessee Valley Authority

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EVALUATOR/CONTACT: P. Lam

SUBJECT: FAILURE OF AN ISOLATION VALVE OF THE REACTOR CORE ISOLATION COOLING SYSTEM TO OPEN AGAINST OPERATING REACTOR PRESSURE

EVENT DATE: March 21, 1984

SUMMARY

On March 21, 1984, the inboard isolation gate valve on the steam supply line of the reactor core isolation cooling (RCIC) system at Browns Ferry Unit 1 could not be opened with a reactor operating pressure differential across the valve disk. The licensee immediately took actions to equalize the pressure on both sides of the valve disk. After two days the valve was successfully opened. During this period, the plant operated at close to full power, and the high pressure coolant injection (HPCI) system was available. The isolation valve which failed is a 3-inch motor-operated solid disk gate valve manufactured by the Walworth Company. Since the valve and its attached motor operator is located inside the drywell, the licensee plans to examine the valve during the next plant outage to determine the exact cause(s) of the failure.

The safety significance of the event, namely the RCIC system being inoperable for 2 days as a result of the isolation valve failing to open against operating reactor pressure, was assessed to be minor. More specifically, the incremental increase in core melt frequency as a result of the 2-day unavailability of the system was determined to be a small fraction of the total core melt frequency for Browns Ferry Unit 1.

However, preliminary investigations were also conducted to assess the design verification basis for RCIC isolation valve closure during a postulated steam supply line break outside containment. The results of these limited investigations indicate that, for some of the newer plants, the valve vendor may demonstrate valve closure capability of such isolation valves by performing an opening test with full design differential pressure across the valve disk. The failure of the Browns Ferry RCIC isolation valve to open against a full operating differential pressure thus casts doubt on its ability to fully close in the presence of the high steam flow and high differential pressure conditions predicted for a postulated RCIC steam line break outside containment. Current surveillance testing of the valve would not detect such a potential inadequacy. Therefore, it is recommended that these implications be considered for further study at the time when the licensee completes an examination of the valve scheduled to be performed during the next planned outage in March of 1985.

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* This document supports ongoing AEOD and NRC activities and does not represent the position or requirements of the responsible NRC program office.

DISCUSSION

1. Event Description

On March 20, 1984, with the plant operating at 96% power, the RCIC system at Browns Ferry Nuclear Plant Unit 1 was removed from service to initiate a repair of a steam leak on a 1-inch drain line between the steam supply line and the main condenser. Outboard containment isolation valve FCV-71-3 on the 3-inch RCIC steam supply line was closed to isolate the steam leak. One day later the steam leak was successfully repaired. Plant personnel then attempted to place the RCIC system back into service without causing a system water hammer. To accomplish this, plant personnel first remotely closed inboard isolation valve FCV-71-2 on the RCIC main steam supply line from the control room panel. Outboard isolation valve FCV-71-3, which had been closed the day before, was then re-opened to permit drainage of any condensate which might have accumulated behind the closed isolation valve while the 1-inch drain valve was being repaired. An attempt was then made to remotely throttle open isolation valve FCV-71-2 which had been closed moments before. Throttling isolation valve FCV-71-2 would allow the isolated RCIC steam supply line piping to be slowly pressurized and warmed up. Plant personnel discovered, however, that inboard isolation valve FCV-71-2 could not be reopened. Following their review, plant personnel concluded that the valve could not be opened because a full reactor operating pressure differential existed across the valve disk.

Since the containment was inerted, the closed inboard isolation valve was not readily accessible for manual opening using the hand wheel of the motor operator. The licensee therefore took the following alternative steps to reopen the valve. First, the outboard isolation valve FCV-71-3 was reclosed. Second, with both the inboard and outboard isolation valves closed, the piping between the valves was pressurized with nitrogen gas to about 1100 psig. This was accomplished using an existing 3/4" test line. This resulted in an equalization of the pressures across the disk of inboard isolation valve FCV-71-2. With pressure across the disk equalized, valve FCV-71-2 was then successfully reopened. The outboard valve was then reopened. Before it was returned to service, valve FCV-71-2 was tested and verified to meet the stroke time requirement of 15 seconds according to the plant technical specifications. However, the valve was not subsequently verified as being able to open with a full differential pressure across the disk. In all, the valve remained closed for 2 days before it was successfully opened (Ref. 1). This event resulted in the RCIC system being unavailable for 2 days. During this period, the reactor operated at close to full power and the HPCI system was available.

The long term corrective action planned by the licensee is to inspect the valve during the next short or refueling outage to determine, if possible, the root cause(s) of failure. The valve is situated in the drywell of the primary containment, hence it can be inspected only when the containment is not inerted.

2. Analysis of Potential Causes of Valve Failure to Open

Until a planned inspection of the isolation valve FCV-71-2 is conducted in the next plant outage, the root cause of its failure to open against operating reactor pressure cannot be determined with certainty. Therefore, discussions here focus on the potential and probable causes of the valve failure.

What is known is that the valve could not be opened against operating reactor pressure and that it was successfully opened once the pressures across its disk were equalized. Furthermore, the licensee stated that limit switches rather than torque switches are used in the valve motor operator control circuitry for isolation valve FCV-71-2. A thermal overload was observed during the initial attempt to open the valve.

Isolation valve FCV-71-2 is a 3-inch motor-operated solid disk gate valve manufactured by the Walworth Company. The valve was designed to operate against a maximum differential pressure of 1200 psi at 550°F (Ref. 2). During the event the maximum valve differential pressure and temperature, respectively, was estimated to be less than 1100 psi and 550°F.

Based on the above information, the potential cause(s) of the valve failure to open against operating reactor pressure can be postulated to be any one of the following mechanisms or a combination of them:

1. Excessive force may have been applied during the valve closure. For example, this could be a result of faulty limit switch or incorrect settings, or distorted valve configurations (see Refs. 3 and 4 for further discussions).

2. Potential thermal binding of the valve disk in its seat following a possible cooldown of the valve body while the valve was closed.

3. Possible failure associated with equipment aging.

4. Possible accumulation of dirt or corrosion of valve parts.

5. Potential inadequacy in design, installation, or maintenance.

6. Other causes yet to be identified.

3. Safety Significance

The failure of isolation valve FCV-71-2 to open is usually not a significant safety concern because it is a normally open valve. For this event, the failure of the valve to open against operating reactor pressure occurred only after it was closed by plant personnel. The additional plant risk associated with isolation valve FCV-71-2 being closed for 2 days was also determined to be minor. The reasons for this assessment are twofold. First, the RCIC system was inoperable for only 2 days as a result of the valve failing to open (during this period the plant operated at close to full power and HPCI was available). The resultant unavailability of the RCIC system on a per-year basis is 2 days/365 days = 6 x 10E-3*, which is about one sixth of the estimated total system unavailability of 4 x 10E-2 from the Interim Reliability Evaluation Program (IREP, Ref. 5) for Browns Ferry Nuclear Plant Unit 1. This increase of RCIC system unavailability is small and within the error band of its associated uncertainties. This is recognized and reflected in the plant technical specifications allowing RCIC to be inoperable for seven days when HPCI is available.

Second, the incremental increase in reactor core melt frequency from such an increase in RCIC system unavailability is determined to be small, of the order of 7 x 10E-7 per reactor year. This incremental increase in core melt frequency is based on re-evaluating, with the estimated increase in RCIC system unavailability, the core melt frequencies associated with the dominant accident sequences for the Browns Ferry Unit 1 plant determined in the IREP study (Table 13, Ref. 5). The total core melt frequency calculated for Browns Ferry Unit 1 in IREP was 2 x 10E-4 per reactor year.

Qualitatively, this conclusion can be obtained from the observation that redundant and diverse means of coolant injection are available in addition to RCIC (namely HPCI; and the low pressure coolant injection and core spray systems in conjunction with the automatic depressurization system) to mitigate accident scenarios in which the reactor pressure remains high. To a lesser extent, the relatively high unavailability of RCIC (about 4 x 10E-2, dominated by the failure of the rupture disks on the RCIC turbine exhaust line) and its low flow capacity (about 600 gpm) also contribute to the assessment that a small increase in RCIC unavailability of the order of 6 x 10E-3 would not significantly increase the reactor core melt frequency, hence reactor accident risks.

4. Valve Closure against High Steam Flow Conditions

A related issue which can be raised by this event is the capability of the isolation valve FCV-71-2, once open, to close against high steam flow conditions during a RCIC steam line break accident. Such an ability would appear to be in doubt because the valve failed to open against operating reactor pressure, which is a condition for which the attached motor operator has been apparently sized. This is based on the following considerations. A discussion with the valve vendor indicated that the design verification testing of valve closure against high steam flow conditions for valve FCV-71-2, if it was ever conducted, was likely a bench test in which the valve was to be opened against a static pressure. Subsequent to a successful design verification test, an inference would then be made that because the valve had demonstrated its operability in opening against a static pressure, it would close against high steam flow conditions. However, dynamic loadings on the disk associated with accelerating flow conditions during valve closure and the absence of a valve motor hammer blow at the end of valve closure tend to indicate that the valve closure against high steam flow conditions may be more difficult than opening against a static pressure. It should be stated

*10E-3 denotes 10

that there is no regulatory requirement that design verification testing or surveillance testing of RCIC isolation valves should be conducted under high steam flow conditions. If one accepts the validity of such an inference which states that a successful valve opening against a static pressure implies valve operability in closure against high steam flow conditions, then valve failure to open against a static pressure would imply that the valve might fail to close against high steam flow conditions.

FINDINGS

1. Isolation valve FCV-71-2 on the RCIC steam supply line could not be opened with a full operating reactor pressure differential across its disk even though the valve is designed to be opened with a larger design differential pressure across the disk. Once the pressures across the valve disk were equalized, which took about 2 days and involved the use of nitrogen in pressurizing the piping between the inboard and outboard isolation valves FCV-71-2 and FCV-71-3, the valve was successfully opened.

2. The event resulted in the RCIC system being unavailable for 2 days during which time the reactor operated at close to full power and the HPCI system was available.

3. The estimated incremental increase in reactor core melt frequency is about 7 x 10E-7 per reactor year which is a small fraction of the total core melt frequency of 2 x 10E-4 per reactor year calculated in the IREP study for Browns Ferry Unit 1.

4. The failure of isolation valve FCV-71-2 to open against operating reactor pressure casts some doubt on its ability to close fully against the high steam flow and differential pressure conditions in a postulated RCIC steam line break accident outside containment. Such a closure capability is a required safety function of the valve. Current surveillance testing of the valve, which is conducted during cold shutdown or refueling, would not detect such a potential inadequacy.

CONCLUSION

The precise cause(s) of isolation valve FCV-71-2 failing to open against a full operating reactor pressure cannot be determined with certainty until a planned inspection is conducted during the next plant outage. The next scheduled plant outage will be in March, 1985. The licensee's findings at that time will be monitored. However, the impact on reactor accident risks of the event, namely the RCIC system being inoperable for 2 days as a result of the isolation valve failure, was assessed to be minor. More specifically, the incremental increase in core melt frequency was determined to be a small fraction of the total core melt frequency assessed in the IREP study.

Finally, the failure of isolation valve FCV-71-2 to open against a full operating reactor pressure casts some doubt on its ability to perform the required function of closing against high steam flow and high differential pressure conditions during a RCIC steam line break accident outside containment. There are apparently no regulatory requirements that design verification testing of the RCIC isolation valve be performed by the valve vendor or that surveillance testing of valve operability be performed by the licensee for high differential pressure conditions. Accordingly, such an inadequacy may remain undetected should it exist. It is suggested, therefore, that a followup AEOD study be considered to further assess these implications and their associated potential safety significance after the licensee examines the valve during the next plant outage.

REFERENCES

1. Licensee Event Report 84-018, Docket 50-259, Browns Ferry Nuclear Plant Unit 1, April 3, 1984.

2. Walworth Company, Cast Steel Pressure Seal Gate Valve, drawing A-12323-M-1-D, 1969.

3. "Inoperable Motor Operated Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment," AEOD/E305, April 13, 1983.

4. "Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980," AEOD/C203, May, 1982.

5. "Interim Reliability Evaluation Program - Analysis of the Browns Ferry, Unit 1, Nuclear Plant," NUREG/CR-2802, July, 1982.