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Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Evaluation of Reactor Building Coolers-Service Water Isolation Valve

Gentlemen:

In our telephone conversations of March 1 and March 2, 1990, we discussed with the NRC Headquarters and Regional Staffs, ANO's actions to repair a leaking reactor building service water coil and the options we were exploring to address leakage of CV-3815 (service water isolation valve from reactor building coolers). Since those calls, ANO has determined that our most appropriate action is to return the affected service water loop to service by removing from service the coil which is leaking and evaluating the impact and safety significance of the CV-3815 leakage. An Engineering evaluation supporting this position is attached. The recommendations included in Section VI of this evaluation will be tracked as action items by ANO. In addition, appropriate compensatory measures will be taken to assure the integrity of the service water loops inside the Reactor Building.

Prior to restart of the unit, the air accumulators for CV-3814 and CV-3815 will be tested to determine how long they can maintain the valves closed following the loss of instrument air pressure (maximum test duration - 6 hours). This test is intended to provide futher assurance that sufficiant time would be available to manually transfer the instrument air compressors to the ES busses following a loss of power during an accident scenario requiring these valves to be closed. Followup actions will include the development of a work plan detailing the installation of jacking devices on CV-3814 and CV-3815 to manually close and maintain the valves closed, if required. This work plan will be referenced in the appropriate abnormal operating procedures and annunciator responses.

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An Engineering study, currently in progress will evaluate the design basis of CV-3814 and CV-3815, and will make recommendations for any design upgrades considered appropriate. The study was commissioned earlier this year and will factor in recent information as well as Generic Letter 88-14 issues, as we have discussed.

Should you have any questions regarding this evaluation, please do not hesitate to contact us.

Very truly yours,

James J. Fisicaro

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ENGINEERING REPORT 90R-1006-01

I. Content of Evaluation/Synopsis

This Engineering Evaluation reviews the background and safety significance of a breach in the Loop II service water piping within the ANO-1 reactor building. It considers the known leakage of this loop's downstream containment isolation valve. The failure is a passive failure of the Reactor Building service water cooling coils or associated service water piping with subsequent leakage of the outboard containment isolation valve(s) on the return lines of the system. Valves of concern are CV-3814 and CV-3815, Contramatic butterfly valves. The safety significance of the potential condition is related to pressurization of the piping system by Reactor Building atmosphere due to post-LOCA conditions. Subsequent leakage of the valves is possible, and offsite dose consequences are considered.

The background of the potential condition developed from a series of actions following a revision to the service water system valve lineup affecting CV-3814 and CV-3815. The outboard containment valves CV-3812/3813 were opened to prevent suspected thermal lockup due to a "bottled up" section of piping between the supply and return valves on the service water loops to each train of Reactor Building coolers. Subsequently, a leak was detected, (via RB sump detection) and found to be a 0.1 gpm leak on VCC-2D, on one of eight separate cooler coil bundles. A desire to isolate the failed bundle via insertion of blind flanges on the coil inlet and outlet led to a consideration of containment integrity since the Service Water piping system would be open to building environment during the repair/maintenance evolution.

In reviewing the status of containment isolation, it was determined that CV-3815 was not leak tight. An alternative to reliance on the valve was developed. A blind flange was installed in a flow instrument location in the Service Water piping inboard of CV-3815 inside the Reactor Building. The flange was tested at about 60 psig to insure building integrity and verified leak tight.

This evaluation addresses the concern of CV-3815 leakage under conditions previously described. No significant safety concern exists with respect to the leakage through the valve considering the design and licensing basis of the Unit. Specific bases for this conclusion with supporting references are contained in the following sections of this evaluation.

II. Design Requirements

A. Design Basis

The two service water loops feeding the reactor building air coolers are closed systems and CV-3815 is the exterior containment isolation valve for loop II which is capable of remote manual operation.

The safety function of the reactor building air coolers and associated service water loops is to provide post accident heat removal capability. Consequently, the primary safety function of the respective service water containment isolation valves, including CV-3815, is to open to permit service water flow to the coolers. These valves receive an automatic open signal from the Engineered Safety Features Actuation System and should remain open for the duration of postulated accidents. In the event of a passive failure of the service water system within the reactor building, the containment building isolation valves are provided to isolate the containment atmosphere from the environment.

10CFR50 Appendix A General Design Criterion 57 provides design criteria for the configuration of containment penetrations of "closed" systems. GDC 57 specifically states:

> "Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve."

The control room operator is capable of remotely closing these valves in the event of a high radiation condition in the service water discharge line. This condition, however, would require the passive failure of the closed service water system inside containment. As noted in II.B.1 below, passive failures of components such as the service water piping or air cooler coils, were not typically considered in the analysis of events for nuclear power facilities of the vintage of ANO-1.

B. Additional Considerations

It should be noted that the stated regulatory requirements noted above are not inconsistent with several other Licensing considerations relative to ANO-1.

1. Design Basis Accidents - Safety Analysis

As covered in the ANO-1 Safety Analysis Reports, design basis accidents were analyzed consistent with standard requirements in place at that time. Specifically, discrete initiating events in combination with credible single failures were analyzed. Further, as evidenced by examination of the analyzed accidents, the credible single failures were basically limited to active failures. Specifically, failures of passive components were considered only as initiating events (i.e. LOCA, MSLB, etc.).

2. Regulatory Requirements

In addition to the specific stipulations of GDC 57 noted in II.A above, it is noted that specific leak rate criteria (i.e. for testing) are not required by 10CFR50 Appendix J as it is applied to ANO. This is consistent insofar as the primary barriers for a closed system are passive components by nature, and other controls (specifically, ILRT testing as noted below) provide additional assurance of overall containment integrity.

3. ILRT/LLRT Technical Specification Controls

A review of ANO-1 technical specification 4.4.1 demonstrates that these surveillances will provide assurance of containment integrity with respect to isolation valves. For example, the ILRT would detect significant leakage through a (failed) closed system and isolation valve. Second, the scope of LLRT testing is intended to cover those components (including isolation valves) that are of primary concern with respect to containment integrity (i.e. specifications 4.4.1.2.1 e, f, and g). CV-3815 does not fall within these criteria as shown below.

a. Spec. 4.4.1.2.1(e) covers valves which provide a direct connection with the reactor building atmosphere; therefore, it is not applicable to a closed system.

b. Spec 4.4.1.2.1 (f) refers to isolation valves whose failure would extend the boundary of the leakage-limiting barrier covered by the ILRT. Since that barrier is provided by the closed system piping, etc. inside containment, it does not apply to CV-3815.

c. Spec 4.4.1.2.1 (g) refers to valves which under post-accident conditions are required to close following the termination of the safety function. Since these valves' safety function is to open and stay open for reactor building cooling, this specification is not applicable.

4. In-Service Testing Program

The ASME Section XI In-Service Testing Program for ANO-1 did not identify CV-3815 as a valve requiring special leak rate testing. The program does not consider the valve to have a safety function of being leak tight. It is our understanding that this is consistent with industry practice relative to "closed system" isolation valves, and is supported by the same considerations noted in II.B.2 above.

III. Safety Implications/Assessment

A. Notwithstanding the above arguments, if a single passive failure is assumed for ANO-1 using criteria similar to that which was used for ANO-2, valve CV-3815 would not be required to close following a DBA. As indicated in the ANO-2 SER Section 15.4.6 of Supplement 2, the worst passive failure considered with respect to offsite dose was a 50 gpm pump seal leak. This leak was required to be postulated 24 hours after the accident. Gross ruptures of pipes were not considered credible passive failures (1). Therefore, if a passive failure of the containment coolers is assumed 24 hours after a DBA accident, a potential release path outside containment would exist.

The potential leak path indicated above will allow the containment atmosphere to flow out this path only if the service water is lost or the service water pressure is below the containment pressure. If a single passive failure of the containment coolers is assumed, then, no other failures are required to be postulated and both service water loops are credited as available. With the service water system available, the back pressure at the containment coolers is projected to be above 15 psig with two loop flow to the lake. The containment back pressure at 24 hours following a DBA is below 7.0 psig (reference 2). Based on the above information, it is apparent that no driving head would exist for the containment atmosphere to travel through this potential flow path. Consequently, the leak tight capability of the isolation valve would not be required even if the ANO-2 passive failure criterion were applied to ANO-1.

B. A further consideration regarding these potential scenarios is an assessment of relative "risk." In order for the containment atmosphere to exit through the containment cooling service water coils, a LOCA would have to occur followed by a coil/tube rupture and a loss of the affected service water loop. The LOCA represents a break of the RCS which potentially allows the containment atmosphere to become contaminated. A service water coil/tube rupture is then required to allow the atmosphere to travel into the service water piping and exit through the leaking valve. The probability of these events occurring in tandem is exceedingly small.

The frequency of a large LOCA is on the order of 8.7×10^{-5} per reactor year based on the ANO-1 IREP data (reference 3). The failure probability of one service water loop has been preliminarily estimated as 0.1 from the work ongoing for the IPE submittal, and the probability of a containment cooler tube rupture over a 30 day mission time is on the order of 3.8×10^{-2} (reference 4). Hence, the overall frequency of dose consequences due to the above identified scenario is 3.3×10^{-7} per reactor year. This is below the value 1.0×10^{-6} used in the IPE submittal for the screening of significant core melt frequencies.

It should also be noted that the above probability is conservatively based on a sequence that will not result in core melt. Consequently, realistic source terms will be very small and the dose implications of the above scenario would be minimal. If the probability of an actual core melt sequence is coupled with the above probability, it is evident that the actual risks are even further diminished.

IV. Interim Measures to Assure the Integrity of Service Water Loop II Within the Reactor Building

Recognizing that the downstream containment isolation valve for Service Water Loop II (CV-3815) is leaking, it is essential that the passive boundary between the containment atmosphere and the environment provided by the service water piping and cooler coils be maintained. Until CV-3815 is repaired, a monitoring program will be in place to ensure this passive boundary. This program will consist of the following elements:

A. The inside containment service water piping will be continually pressurized by the service water system to aid in leak detection.

B. The reactor building sump levels are checked and logged each shift.

C. If readings from the reactor building sump indicate changes in leakage within the building, samples of the sump liquid will be analyzed to determine if the leakage could be from the Service Water System.

D. If it is determined that the Service Water System could possibly be leaking within the reactor building, a building entry will be made to determine the source of the leakage.

E. If during the course of operations, leakage is identified in the reactor building portion of the Service Water Loop II, those actions specified by Technical Specification 3.6. will be taken without credit for CV-3815.

F. The coolers will be tested at elevated pressure prior to restart in order to verify overall system integrity.

V. Conclusion

Based on the above information, we believe adequate justification is provided to support operation of ANO-1 from the present until the upcoming refueling outage.

The design and testing requirements of this valve do not consider any specific criteria for valve leakage. Consequently, the current condition is considered to be in compliance with the existing design bases. The safety implications of this condition are minimal considering realistic risk factors. Interim measures to assure the integrity of the closed piping in containment have been provided, further minimizing the significance of the leaking valve. The assessment reflects design and licensing requirements consistent with plants of the ANO-1 vintage and provides adequate assurance that public health and safety protective measures have not been reduced.

VI. Recommendations

Although the design/licensing basis of the Unit does not require rigorous testing and leakage acceptance criteria for these valves, good engineering practice is to acknowledge the functional requirements of the system and address those requirements with operations/maintenance programs including surveillance. The following recommendations acknowledge the containment isolation function of the valves, and represent good practice even though the specific design/licensing bases do not mandate the items.

A. Survey other utility practices in leak testing, surveillance criteria, and App. J. and Section XI program interpretations for this class of valves.

B. Determine if repair or maintenance can bring the valves within manufacturers criteria for leakage and overall performance.

C. Evaluate replacement of the valves with a different type with better leakage performance. This evaluation will be supported by the Item 1. survey and the Item 2. maintenance review.

D. Establish a position on surveillance and testing of the valves to address the containment isolation function. Add requirements to the Section XI program.

E. Review all GDC 57 valves to determine the need for upgraded or augmented testing and surveillance.

These recommendations are short-term actions to be completed by the 1R9 outage. Although replacement of the valves may not be feasible by 1R9, if not replaced, the valves should have known performance and condition established so that surveillance and testing can be benchmarked.

VII. References

1. NRC Information Report, SECY-77-439, Subject: "Single Failure Criteria", August 17, 1977.

2. AP&L Calculation 88E-0098-10, "DBA LOCA w/1600 gpm Service Water Flow @ 95°F, Smaller RCB, Smaller BWST, Additional Heat Input," Rev 0.

3. NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One - Unit 1 Nuclear Power Plant," Volume 1, June 1982.

4. Science Applications International Corporation, "Generic Data Notebook for Commercial Nuclear Power Plant Probabilistic Risk Assessment," Martin Stutzke, May 1989.