UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO IN-PLANT SAFETY RELIEF VALVE DISCHARGE TEST RESULTS

LICENSE CONDITION 2.C.(4)

DETROIT EDISON COMPANY

WOLVERINE POWER SUPPLY COOPERATIVE, INCORPORATED

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

In 1975, the NRC identified concerns regarding Boiling Water Reactor (BWR) Mark I containment system design. These concerns were based upon test results showing increased dynamic loads from safety relief valve (SRV) discharges. The SRVs are mounted on the main steam lines inside the dry well, with discharge pipes routed into the suppression pool. The scenario was postulated as follows: When an SRV is actuated to provide overpressure protection for a primary system, steam is discharged from the primary system through the SRV into a discharge line that leads to the pressure suppression pool. Air initially exists in the line and is compressed by the influx of steam. The water column at the end of the line which is submerged in the pool is expelled first through T-Quenchers mounted at the submerged end of the line. This water column is followed by the compressed air, which forms one or more air bubbles in the pool. Each bubble undergoes oscillatory expansion and contractions as it rises to the surface of the pool. Following the air-clearing phase, steam is injected into the pool through the quencher. The steam-water interfaces formed at the quencher during this phase is stable as long as the local pool temperature remains below the normal boiling temperature of 212°F. In summary, the discharge of both the air, which was in the SRV line, and the steam into the suppression pool produces hydrodynamic loads on the containment structure, piping, and equipment.

The key parameters affecting the loads and the pool temperature gradients have been identified through generic testing. However, concerns have been expressed that there is enough uncertainly about the interdependence and quantitative effects of plant-specific variables that confirmatory testing should be conducted in plants in which these parameters are substantially different from those previously tested. The generic approach was accepted by the NRC in NUREG-0661, "Safety Evaluation Report for Mark I Containment Long-Term Program," dated July 1980 and in Supplement No. 1 dated August 1982. The guidelines for the in-plant tests were provided in NUREG-0763, "Guidelines for Confirmatory In-plant Tests of Safety Relief Valve Discharge for BWR Plants," dated May 1981. The generic analysis was to be applied to each Mark I containment plant through the use of a Plant Unique Analysis (PUA).

This process was undertaken for the Enrico Fermi Atomic Power Plant Unit 2 (Fermi-2) as part of the initial plant licensing. The NRC accepted the Fermi-2 PUA contingent upon confirmation by in-plant test of the conservatism of the load reduction factor used in the calculation of SRV water jet impingement and air bubble drag loads. The Fermi-2 Operating License included License Condition 2.C.(4) requiring these in-plant tests and submission of an analysis of the test results to the NRC staff within six months of the completion of the testing.

2.0 BACKGROUND

In order to satisfy the licensing condition, Detroit Edison conducted a series of in-plant SRV discharge tests for Fermi-2 on March 12, 1987. The test was preceded by shakedown tests performed on March 11, 1987. The test matrix consisted of two shakedown tests, four single valve actuations (SVA), and four consecutive valve actuations (CVA). The tests followed the general guidelines provided in NUREG-0661 and NUREG-0763.

The test program focused on measurement of the following:

1. Peak suppression pool boundary pressures during SRV discharge line (SRVDL) air clearing and steam discharge due to a single SRV actuation under normal water level in the submerged section of the discharge line and under both cold and hot conditions of the line.

2. Pressure magnitude and frequency content of the T-Quencher air bubble pressure transients.

3. Water and air clearing reaction loads on the SRVDL and T-Quencher supports.

4. Suppression chamber structural response including torus shell membrane stresses due to a single SRV discharge (cold and hot pipe).

Four types of instrumentation were used for sensing and measuring the parameters. The instruments are pressure transducers, stain gauges, accelerometers, resistance temperature detectors and existing plant system thermocouples. Twenty pressure transducers were installed to measure the torus shell, SRVDL and T-Quencher air bubble and internal pressures. Ninety-eight stain gauges were installed on the containment, submerged structures and piping to measure representative strain data. Four accelerometers were installed to measure the torus shell and torus-attached piping response to SRV discharge loads. Two temperature sensors were used to monitor the SRVDL vent line penetration and wet well SRVDL temperature.

Upon completion of the test program, Detroit Edison submitted to the NRC a report dated November 13, 1987 describing the test procedures, test instrumentation, and the results as discussed above. The report entitled "Final Test Report: In-Plant Safety Relief Valve Discharge Test - Enrico Fermi Atomic Power Plant, Unit 2" was prepared by Nutech Engineers Inc.

After a preliminary review of the above report, the NRC requested additional information and clarification in the following areas:

1. Sequence of testing
2. Testing temperature
3. 90-90 vs 95-95 statistical results
4. Analytical results used for comparison with test data

Detroit Edison provided the requested information by letter dated August 12, 1988.

3.0 EVALUATION

The NRC staff has completed review of the report and the additional information. The following is a summary of the evaluation.

Regarding testing sequence and temperature, the scope of the Fermi-2 in-plant test was limited to confirmation of the SRV discharge methodology used in the Fermi PUA, and this was mainly to address the issue that the discharge configuration at Fermi-2 is geometrically different from configurations tested previously. Suppression pool thermal mixing tests were not performed since the pool temperature response to SRV transients described in the PUA demonstrates compliance with the required pool temperature limits.

NUREG-0763 recommends testing under normal discharge conditions and that plant specific tests generally not include leaking valve actuations (LVA). For inadvertent testing under LVA conditions, load changes should be quantified on a generic basis. For Fermi-2 all but one of the SVA discharge tests were performed with a tail pipe temperature of approximately 212°F which is above the range of normal plant operation temperature. This was indicative of a leaking valve. Compared to the one test performed with the ambient temperature, the SRV tests under the leaking valve condition tend to increase the dominant bubble frequency toward the fundamental frequency of the suppression pool chamber and results in a conservative shift of the discharge loads. Therefore, the elevated tail pipe temperature conditions are acceptable.

Regarding the test and analysis results, the test data were statistically analyzed to obtain a 90-90 probability value which was then compared with the results obtained in the PUA. In response to a request for 95-95 probability values, the licensee concluded that the quality of the test data is such that the magnitude of the change (90-90 vs 95-95) would be small. Margins in the PUA results compared to the test 90-90 data are appreciably high. Therefore, 95-95 values are expected to be bounded by the PUA results.

The critical issues being addressed are whether the tests were sequentially and properly performed and the proper parameters characterizing the actual SRV hydrodynamic loading were accurately measured and compared to the analytically estimated values used in the PUA reports. This is to be done within the guidelines of NUREG-0661 and NUREG-0763. The key parameters selected for this purpose and the results of the comparison are as follows:

1. Peak Pressure - The measured peak T-Quencher bubble pressure and the torus shell pressure are less than 50% of predicted values. The test data were analyzed for a 90-90 probability value, i.e., 90% confidence that 90% of measured results will be less than the peak pressure. T-Quencher bubble frequencies showed good correlation with the predicted values.

2. Reaction loads - The measured water and air clearing reaction loads on the SRV discharge line and T-Quencher supports were about 20% of the analytically predicted values and test conditions.

3. Strains - The measured strains on the torus shell, torus support structures, internal structures, and piping compared favorably with the predicted values (about 10%-40% of the analytically predicted values).

4. Zero Period Acceleration (ZPA) - The peak measured zero period acceleration is well below the analytically predicted response at each location. In addition, the measured coupled system acceleration (torus shell/piping systems) are less than 50% of the analyzed clean shell response.

4.0 CONCLUSIONS

Detroit Edison has satisfied the license condition of performing an in-plant test for confirming the SRV discharge loads in accordance with the guidelines provided in NUREG-0661 and NUREG-0763.

Based on the licensee's submittals, the NRC staff concludes that the Fermi-2 plant unique analysis loads are conservative and the safety margin in the design of the primary containment system for SRV discharge loads is adequate.

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