Omaha Public Power District

1623 HARNEY ■ OMAHA, NEBRASKA 68102 ■ TELEPHONE 536-4000 AREA CODE 402

December 30, 1982
LIC-82-415

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

NUREG-0737, Item II.D.1
Relief and Safety Valve Test Program

Omaha Public Power District's letter to the Commission dated July 1, 1982 provided the District's latest report in response to the subject TMI Action Plan item. In this report, the District also identified two items for which further information and evaluation were necessary. These items involved: (1) plant specific evaluation of the operability of the Fort Calhoun Station pressurizer safety valves under various conditions and (2) evaluation of the effect of thermal and dynamic stresses on the power operated relief valve/safety valve (R/SV) piping system and supports. The District has completed our analysis of these items and the results are provided in the attachment.

As a result of the District's and Combustion Engineering's analyses, it has been determined that the safety valve and inlet piping failures are bounded by existing safety analyses. However, modifications are being considered to improve valve and inlet piping operability since proper operation is considered highly desirable. The District's analysis of the S/RV discharge piping indicates that significant stresses can be exerted upon actuation of these valves. The District will ensure that continued safe operation is assured prior to restart of the Fort Calhoun Station following the refueling outage presently in progress.

Sincerely,

W. C. Jones
Division Manager
Production Operations

A046

WCJ/TLP:jmm

Attachment

cc: LeBoeuf, Lamb, Leiby & MacRae

ATTACHMENT

References 1) and 2) addressed the issue of power operated relief valve (PORV) operability and concluded that the PORV's would function adequately for the conditions applicable to the Fort Calhoun Station. Therefore, this report only addresses safety valve (SV) operability and the impact of safety/relief valve (S/RV) piping system response on S/RV performance and safety-related equipment integrity.

The bounding cases for S/RV system malfunction are:

1. Failure of the S/RV's to open with pressure increasing above the valves' setpoints.

2. Prolonged blowdown due to failure of the valves to close on demand.

3. Failure of the S/RV piping system under the action of thermal and dynamic loading.

With regard to case 1, the Fort Calhoun Station is equipped with two PORV's and two SV's which function independently. Thus, the District believes there is a high probability that at least one valve would open at or near its pressure setpoint and, as previously detailed, the PORV's should operate for all conditions applicable to the Fort Calhoun Station. Additionally, the reactor coolant system (RCS) was analyzed (Reference 3) by our Nuclear Steam System Supply (NSSS) vendor (i.e., Combustion Engineering) at various S/RV setpoint capacities to determine the resultant RCS pressure vs. time history. The assumed transient used for this S/RV analysis was a conservative version of the loss of load transient (LOLT), as identified in the Fort Calhoun Station's Updated Safety Analysis Report (USAR), Section 14.9. As detailed in Reference 3), the S/RV analysis demonstrated that RCS pressure is self-limiting and pressure would remain below the code limit of 2750 psia without reliance on the S/RV system (i.e., a 0.0% S/RV capacity). Thus, the S/RV analysis demonstrated that SV operability is not required to ensure plant safety. However, the District believes it is desirable that the SV's perform satisfactorily if actuated and will initiate measures as detailed below to achieve satisfactory performance.

EPRI's S/RV performance tests identified a number of factors which had significant impact on SV performance. The factors applicable to the Fort Calhoun Station are:

1. Large inlet pressure drops upon valve opening.

2. Ring position adjustment.

3. Inlet pressure ramp rate at valve actuation.

4. Loop seal water inlet piping.

Factors 1, 2, and 3 were evaluated by Combustion Engineering for the District and the conclusion is that stable valve performance, on steam, can be achieved with appropriate ring settings and inlet piping configurations. This analysis and conclusions, which are detailed in Reference 4), have been forwarded to the Commission via the CE Owners Group. Modifications and/or adjustments recommended by CE for ensuring stable valve performance are presently being evaluated and any modifications or adjustments needed to ensure proper valve stability will be completed during the first refueling outage after required materials and/or approved designs are available.

Factor 4 has been evaluated and the resulting recommendations for ensuring stable valve performance are consistent with those of factors 1, 2, and 3. The District will retain filled loop seal inlets on our S/RV's, but we have concluded that SV performance can be improved by reducing the volume and increasing the temperature of the loop seal water. These modifications are presently being evaluated and, if needed, will be completed once final designs and equipment are available. An added benefit to be gained by instituting these modifications will be realized by reducing the dynamic loading for the piping system.

Case 2, extended blowdown due to failure of valves to close, has been evaluated at two levels of severity. The most severe level, that of one or more valves failing to close, will be addressed with case 3. The least severe, that of the valves closing but allowing blowdowns in excess of 5%, is addressed separately. The applicable design code for the SV's requires justification for SV blowdowns in excess of 5%. EPRI testing, the results of which are published in Volume 5 of the interim report (July, 1982) of EPRI Research Project V102-2, revealed that the adjustment ring settings which resulted in stable SV performance, for typical pressurizer S/RV systems, resulted in blowdowns which were greater than 5%. Our NSSS vendor performed an analysis to determine the effects on the RCS system of blowdowns to 20% for the LOLT. The results of the analysis which are given in Reference 4) show that blowdowns up to 20% are acceptable (i.e., "discharge through the valves is limited to steam, while RCS loop subcooling is maintained throughout the associated blowdown"). The blowdown for the recommended stable ring settings at the Fort Calhoun Station is not expected to exceed 20%.

Case 3, failure of the S/RV inlet piping system under the action of thermal and dynamic loading, is assumed to result in a loss of reactor coolant into containment, either as a result of a stuck open S/RV or a pipe break. The most severe event would occur if all the S/RV inlet pipes were to suffer guillotine breaks at the pressurizer nozzle connections. A review of the Fort Calhoun Station loss of coolant accident (LOCA) analysis has established that the consequences of the resultant fluid release would not prevent the safe shutdown of the reactor. Since the total cross sectional area of the three nozzle connections is less than 0.5 ft^2, they are covered by the small break LOCA analysis, USAR Section 14.15.2.2.

An analysis of the Fort Calhoun Station S/RV inlet piping system under the action of dynamic loads in combination with normal operating

loads, has revealed that elastic limits could be exceeded and plastic deformation of the SV inlet piping may occur during loop seal discharges unless pipe/restraint modifications are made. However, this is not a probable event since PORV actuation will preclude SV actuation, as demonstrated in Reference 3. Additionally, modifications to the loop seal configuration are being considered that would result in reduced loading of the inlet piping and concurrent restraint modifications will be included if it is deemed necessary to perform such modifications. Analysis of the PORV inlet piping shows stresses to be within applicable code allowables for the events analyzed. In addition, PORV flange loads are less than those measured in EPRI tests and, therefore, the valves are not expected to stick open (reference EPRI Test #16-DR-6W, May 8, 1981, EPRI/PWR Safety and Relief Valve Test Summary, Report #4069). A LOCA due to S/RV inlet piping failure is therefore unlikely.

An analysis of the S/RV discharge piping indicated stresses and restraint loads downstream of the S/RV's are in excess of design values. The District will continue to evaluate the implications of this analysis finding during the present refueling outage and will ensure that continued safe operation of the plant is justified prior to restart. It should be noted that the District has experienced a PORV actuation in the past without any degradation of the discharge piping.

REFERENCES

1) Letter from W. C. Jones (OPPD) to Robert A. Clark (NRC) dated July 1, 1982.

2) CEN-213, "Summary Report on the Operability of Power Operated Relief Valves in CE Designed Plants".

3) Combustion Engineering letter, SE-82-837, to the Safety and Relief Valve Subcommittee (CEOG) dated December 22, 1982.

4) CEN-227, "Summary Report on the Operability of Pressurizer Safety Valves in CE Designed Plants".