



TOLEDO EDISON COMPANY  
DAVIS-BESSE NUCLEAR POWER STATION UNIT ONE  
SUPPLEMENTAL INFORMATION FOR LER NP-32-77-12

DATE OF EVENT: August 5, 1977

FACILITY: Davis-Besse Unit 1

IDENTIFICATION OF OCCURRENCE: Reduction of Allowable Pressurizer Relief Actuations

Conditions Prior to Occurrence: The unit was in Mode 3, with Power (MWT) = 0, and Load (Gross MWE) = 0.

Description of Occurrence: During Hot Functional Testing, the three inch pressurizer power operated relief valve (PORV) was opened and severe movement of the eight inch discharge piping was observed. The slope in the inlet piping to the PORV acts as a loop seal which permits two phase flow to be initially discharged when the relief lifts. Both pressurizer code safety valves have actual loop seals designed into the inlet piping to prevent the valve seating surfaces from being exposed to a high temperature steam environment. This was to prevent leakage problems caused by continuous exposure to a potentially corrosive steam atmosphere.

Analysis of the problem indicated that maintaining 600°F on the upstream portion of the three inch line, and 550°F on the loop seals of the six inch safety valve lines would permit the water to flash to steam upon valve discharge. This would considerably limit the discharge piping movement. The design objectives for the permanent electric heater have not been met requiring that the number of allowable lifetime valve actuations be significantly reduced.

This is reportable per Technical Specification 6.9.1.8.i in that the performance of the structure requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the safety analysis report. The stress analysis conducted in 1977 by a consultant for the architect/engineer excessively reduced the allowable actuations due to the conservatisms of a simplified analysis. Continued review of the situation by the same consultant in 1978 using the life history to date imposed a limit of ten additional thermal transient type lifts at or above 400°F, however a subsequent study increased the limitation to 25 hot lifts based on a more realistic allocation of the usage factor to various load sets. The results of a more exact inelastic stress analysis using both fatigue and thermal ratcheting criteria, completed in late 1978, updated the allowable number of hot and cold lifts to 650 and 25 respectively.

During this ongoing study, it was realized that the original analysis did not account for two of the three stanchions welded to the pipe. These welded attachments, acting as stress risers, became the limiting critical location for the following analyses. Further investigation revealed that one stanchion designed for lateral loads would also restraint torsional motion, which was not factored into the original stress assumptions for the Class 1 piping. Additionally, the three stanchions on the three inch piping have full penetration welds required by the 1971 edition of the ASME Code. However, because of this configuration, these welds cannot be inspected by volumetric examination, and the condition of these welds due to the cycling of these lines over the past several years cannot be determined.

Designation of Apparent Cause of Occurrence: The loop seals leading to the inlet of the pressurizer reliefs allow a water slug to cause severe pipe movement during valve lifts. Permanent strip heater installation on both three inch and six inch lines would not maintain design temperatures to permit the water to flash to steam if the valve was opened, which would considerably reduce discharge piping movement. Welded attachments to the pipe were not properly treated in the original stress analysis, and the condition of their full penetration welds cannot be verified radiographically due to the support's configuration.

Analysis of Occurrence: There was no danger to the health and safety of the public or to station personnel. Valve discharges to date are well below the lifetime cycles.

Corrective Action: During Hot Functional Testing, additional supports were added to the eight inch common discharge line. A permanent electric heater installation was provided along with additional insulation. With greater than 410°F maintained by heat tracing, a lifetime limit of 30 discharges was imposed on the six inch code safety valves compared to a B&W design criteria of 288 actuations. Two lifts at temperatures less than 410°F was permitted. Similarly, thermal stress analysis of the three inch PORV piping established a limit of 500 discharges for a loop seal temperature of 560°F as compared to a B&W design criteria of 8800 lifetime valve lift transients. With the addition of the supports and heaters, the discharge pipe movement was brought into acceptable limits.

Subsequent, more exacting analyses have expanded the PORV allowable operating transient cycles. A Periodic Test, PT 5164.03, "Pressurizer Relief Valve Heat Trace Test" will monitor daily the piping temperatures. Actuations will be logged in the transient file maintained per AD 1839.01, "Documentation of Allowable Operating Transient Cycles". As of this May 19, 1980, the unit has used only 91 of the 650 allowed greater than 400°F actuations and 17 of the 25 < 400°F actuations.

Continued analysis of the stress problems revealed that three stanchions on the three inch piping are critical stress risers which restrict further relaxation on the number of permitted PORV lifts. Discrepancies between the stress analysis assumptions regarding torsional restraint and the actual support configuration was identified on stanchion 30-CCA-8-H1. This welded attachment in addition to 30-CCA-8-H2 and 30-CCA-8-H6 have full penetration welds which cannot be verified volumetrically. Concern over the condition of these welds due to the original pipe movement cannot be validated. During the spring 1980 refueling outage, these supports will be removed and replaced with redesigned hangers under Facility Change Request 80-121. Two of these redesigns will not use any welded (to the pipe) attachments, and the other, an axial restraint, will use welded lugs of a design that will permit volumetric examination of the full penetration weld.

Under Facility Change Request 79-356, all three pressurizer loop seals will be removed. For the PORV, this modification and the stanchion modification would increase the number of allowable cycles back to the original NSSS vendor criteria. Continued stress

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Failure Data: No previous design deficiencies of Reactor Coolant System piping configurations have been reported.