NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

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LICENSEE CODE / LICENSE NUMBER / LICENSE TYPE / CAT

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DOCKET NUMBER / EVENT DATE / REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 (NP-32-77-12) During Hot Functional Testing, the 3" pressurizer power operated re-
0 3 lief valve (PORV) was opened and severe movement of the 8" discharge piping was ob-
0 4 served. Steps were taken to correct the problem and develop a realistic number of
0 5 allowed thermal transient type lifts. The 1982 EPRI valve testing program identified
0 6 that more reliable operation of the code safeties could be achieved by mounting them
0 7 directly on the pressurizer. This is reportable per T.S. 6.9.1.8.i. There was no
0 8 danger to the public or station personnel. The lifts to date are below lifetime cycles

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	C J (11)	B (12)	A (13)	Z Z Z Z Z Z (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
7 7	—	Z Z Z	/	Ø 1	T	—	3

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
F (18)	F (19)	Z (20)	Z (21)	Ø Ø Ø Ø	Y (23)	N (24)	A (25)	Z Z Z Z (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 Potential valve chattering identified by EPRI is due to sonic traveling wave func-
1 1 tions. Loop seals on the relief inlets allowed a water slug to cause severe discharge
1 2 pipe movement upon valve lifts. FCR 82-074 was implemented during the 1982 refueling
1 3 outage to relocate the code safeties to the top of the pressurizer. The new configur-
1 4 ation will discharge into a T connection directly into the containment environment.

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	G (28)	Ø Ø Ø (29)	NA	D (31)	EPRI Testing

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	Ø Ø Ø (37)	Z (38)	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	Ø Ø Ø (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	NA

PUBLICITY

	ISSUED	DESCRIPTION (45)
2 0	N (44)	NA

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DVR 097-1

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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
3 Actuations and Relocation of Pressurizer Code Safety Valves.

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 racheting 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.

In 1982, the EPRI valve testing program along with the Teledyne analyses identified that more reliable operation of the code safeties could be achieved by mounting them directly on the pressurizer. The old configura-
3 tion existing at the start of the 1982 refueling outage is susceptible to valve chattering problems. Studies indicate the valve chatter could be severe enough to cause damage to the relief valves. In addition, the prior existing loop seals present a damage potential.

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.

Potential valve chattering is due to the sonic traveling wave formation which could limit the relieving capacity of these valves. The loop seals were required to protect the valve disc/seat from H_2 cutting. However, when the valve pops open, the liquid loop seal is accelerated towards
3 the valve inlet. The loop seals were therefore heat traced to attempt to keep the loop seal at a temperature that would ensure complete flashing prior to reaching the valve inlet. Impact by liquid at inlet velocities would have an equal potential for valve damage.

Analysis of Occurrence: There was no danger to the health and safety of the public or to station personnel. PORV discharges to date are well below
3 the lifetime cycles. The code safeties have never had to relieve in service.

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 analysis is being performed to reflect the post-TMI, 2400 psig lift setpoint of the PORV.

As a result of the 1982 EPRI study, FCR 82-074 was written and implemented during the 1982 refueling outage. This FCR relocated the Pressurizer Code Safeties to a new location at the top of the pressurizer. The loop seal upstream of the old location will no longer be used. The disc in the valves were replaced with a soft seat material (19-9DL) suitable for
3 operation without the loop seal. The new configuration has the valves discharging into a T connection directly into the containment environment. The lift setting of 2435 psig will not be changed.

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