THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

P.O. BOX 5000 - CLEVELAND, OHIO 44101 - TELEPHONE (216) 622-9800 - ILLUMINATING BLDG. — 55 PUBLIC SQUARE

Serving The Best Location in the Nation

MURRAY R. EDELMAN
VICE PRESIDENT
NUCLEAR

July 8, 1985
PY-CEI/NRR-0282 L

Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Response to Mechanical
Engineering Branch Question 210.9
SRV Piping and Support Analysis

Dear Mr. Younglood:

This letter and its attachment provides our response to the remaining item in your December 14, 1982 request for additional information regarding the safety relief valve piping and quencher device. (Question 210.9 and 210.10). Previous correspondence in response to this subject was provided on December 16, 1982, May 16, 1983 (PY-CEI/NRR-0043L), June 21, 1983 (PY-CEI/NRR-0054L) and on February 28, 1985 (PY-CEI/NRR-0193L), which responded to your follow-up letter dated January 22, 1985.

Attachment 1 to this letter provides our analysis of the welded attachment which verifies the structural integrity of the safety relief valve piping and the quencher support capabilities. (Question 210.9) The detailed analyses have shown that, although classified as ASME Class 3, the piping with the welded attachment satisfies ASME Code Class 1 stress requirements.

We believe that this information completes our responses to all of the Mechanical Engineering Branch questions. If you have any questions, please let me know.

Very truly yours,

Murray R. Edelman

Murray R. Edelman
Vice President
Nuclear Group

MRE:njc

Attachments

cc: Jay Silberg, Esq.
John Stefano (2)
J. Grobe

B001
1/1

RESPONSE TO NRC QUESTION 210.09
SRV PIPE STRESS AT WELDED ATTACHMENT
PERRY PROJECT

THE PURPOSE

The purpose of this report is to document the NRC question, the circumstances, the specific concern as interpreted from the NRC question, the outline of the analysis approach taken to address the NRC staff's concerns and the conclusions drawn from the analysis.

THE NRC QUESTION

Nuclear Regulatory Commission request (Ref. 1), question 210.09, states the following:

> "In recent discussions with the applicant, it has become apparent to the staff that the safety-relief valve (SRV) piping in the suppression pool are provided with horizontal quencher supports that are welded directly to the piping. In the staff's opinion, the welded attachment could cause excessive localized bending stresses and harmful thermal gradients in the piping following actuations of the safety-relief valves. Furthermore, the relatively large number of stress cycles could result in fatigue failure in the piping. The applicant is requested to provide detailed analyses of the welded attachment to verify that the structural integrity of the piping and the quencher support capabilities are not compromised."

THE SYSTEM, THE SUPPORT, AND THE ATTACHMENT

Perry Unit 1 reactor has 19 safety-relief lines for overpressure protection. Each line runs independently from the safety-relief valve to the quencher in the suppression pool (Ref. 2.)

The lower pipe segment of each relief line is supported in the horizontal direction from the Drywell wall by two struts. (Ref. 3.) These struts also function as the horizontal support of the quencher because the lower segment of the relief line piping and the quencher are an integral part of the relief line. The support attachment at the pipe consists of a skewed trunnion welded to a pad which in turn is welded to the relief line pipe. The attachment is submerged in the pool under normal conditions. The pipe is 10" schedule 40S having 0.365" wall made of SA358 TP304 steel. The pad is SA240 TP304 steel 1" thick, 20" long, and wraps around one-half of the pipe circumference.

THE NRC STAFF'S CONCERN

When an SRV is actuated to relieve the pressure in the reactor by discharging the steam through the relief pipe to the suppression pool, the pipe will be heated by the 370°F steam (Ref. 4). The temperature gradient through the pipe wall, the support attachment pad, and the trunnion and the structural discontinuity induce local stresses in the pipe. The NRC staff's concerns are: (1) whether the localized bending stress and the thermal gradient stress in the pipe are excessive and (2) whether the large number of stress cycles could result in fatigue failure in the pipe.

THE ANALYSIS APPROACH OUTLINE

Code Class

Perry Project Piping Design Specification DSP-B21-1-4549-00 Rev. 1 (Ref. 5) specifies the SRV relief piping as ASME Class 3 which shall be designed in accordance with the rules of subsection ND of the ASME Boiler and Pressure Vessel (B&PV) code.

Code Rules for the Analysis

The conventional piping analysis method for Class 3 systems uses ND-3600 "PIPING DESIGN" rules of the B&PV code. These design rules do not specifically address detailed analysis of the thermal gradient stress, the localized bending stress in the pipe at an integral attachment, or the fatigue life of the pipe under cyclic loads. In response to the NRC request, NB-3200 "DESIGN BY ANALYSIS" rules for Class-1 components were used for the evaluation of the stresses and fatigue of the pipe at the welded attachment.

Original Support Design

The original support design used the ANSYS finite element computer program for the analysis of the stresses in the pipe reacting to the support load. ANSYS STIF-63, quadrilateral shell elements were used for the 3-dimensional model, which included the pipe, the pad, and the trunnion. The support reaction forces were distributed over the outer end of the trunnion.

Thermal Transient Analysis

ANSYS computer program was used both for the heat transfer analysis to establish the temperature distribution histories, and for the stress analysis to compute the thermal stresses resulting from the uneven temperature distribution. The pad thickness (1") was divided into 8 uniform elements

and the pipe wall (0.365") was divided into 4 elements. The initial solution time step was 1 second; the transient solution covered 30 minutes. ANSYS 2-dimensional axisymmetrical elements STIF-55 and STIF-42 were used for the thermal transient analysis and the thermal stress analysis, respectively. In this model, the pad was replaced by a full circumferential sleeve. This approach was taken because the existing 3-dimensional structural analysis element does not have an equivalent element for heat transfer analysis. The justification for using a 2-dimensional model instead of a 3-dimensional model is noted in the following section.

Justification of Thermal Stress Analysis Model

The following conservative approaches were used in the thermal stress analysis:

A. The SRV blowdown thermal fluid analysis (Ref. 4) has established the temperature of the steam in the pipe in the vicinity of the welded attachment to be 370° F. A step change of temperature from 70° F to 370° F was used in the thermal transient analysis instead of a less severe ramp temperature change with finite rise time.

B. The number of valve actuations was assumed to be 1,800. The design specification (Ref. 5) specifies 220 occurrences for multiple valve actuation and 1,580 occurrences for single valve actuation. In accordance with the definition of the specification, one-third of the valves will not open more than 220 times in the life of the plant.

C. The maximum thermal stress and the maximum support reaction stress do not occur at the same time at the same location on the pipe in the same relief line. These two stresses were added to obtain the total stress without considering the actual time phase difference for occurrence of the two types of stress.

The 2-dimensional thermal analysis model used does not provide precise localized thermal stresses in the pipe wall off the pad in the circumferential direction. This is because the full circumferential sleeve instead of a pad is

used. However, a separate pad model was used to show that the local thermal stress in the pipe wall off the pad in the circumferential direction is approximately the same as the stress in the pipe wall off the pad in the longitudinal direction (within 10%). Consequently, the highest stress elements chosen from the 2-dimensional model should reasonably cover the worst condition anywhere in the pipe near the pad irrespective of their locations in relation to the orientation of the pad.

The accumulated conservatism from these causes and the stress margins remaining are more than sufficient to account for the discrepancies caused by using a 2-D model instead of a 3-D model.

NB-3200 Code Compliance Evaluation

The stresses calculated from the thermal transient stress analysis, the component structural analysis and the pressure stress were input to G/C in-house program NB3200/M107A for code compliance evaluation in accordance with the NB-3200 rules.

NB-3200P/M107A Computer Program Description (Ref. 6)

The NB-3200P/M107A computer program was developed by G/C for the purpose of evaluating the Class-1 piping components in accordance with the rules of NB-3200 of the ASME B&PV code. The program consists of three main segments. The first segment develops the load combination for each service and design condition in accordance with the Perry piping design specification. The second segment generates the stress amplification matrix. The third segment evaluates the primary stress, the secondary stress, the fatigue usage factor, and plastic-elastic analysis in accordance with NB-3200 rules.

DESIGN DATA

The following data are established in the design specification (Ref. 5):

A.	Internal Pressure	570 psig
B.	Design Temperature	477° F
	(Steam Temperature calculated)	370° F (Ref. 4)
C.	Flow Rate	1 X 10^6 lbs/hr
	(Relief Valve Operation @ Max. Reactor Pressure)	

D. Pipe Material

SA358 TP304 Class 1 Stainless Steel

Code allowable stress at 477° F S_m = 17,700 psi

Maximum Support Reaction Loads per Strut

Deadweight	1,368 lbs.
Thermal Expansion	-12,982 lbs.
OBE	± 535
OBE Displacements	± 109
SSE	± 783
Chugging	± 24,252
Fluid Flow Transient (max)	+ 4,860
Fluid Flow Transient (min)	-708
SRV Pool Drag	± 31,658
SRV Inertia	± 2,400

Design Conditions Used in Analysis

For conservative simplification of the analysis, the highest steam temperature at the support location for all 19 lines and the highest support reaction loads among all 19 lines are assumed to occur at a single attachment. Therefore the results of one analysis covers all 19 lines.

SUMMARY OF RESULTS (Ref. 7, M107 Output B21QNCHNB/J466 P.8)

Design Condition Stresses (psi)

	Analyzed	Allowable
P_m	2,276	S_m = 17,700
$P_L + P_b$	2,755	1.5 S_m = 26,550

Upset Condition Stresses (psi)

	Analyzed	Allowable
P_e	23,471	3 S_m = 53,100
$P_L + P_b + P_e + Q$	48,190	3 S_m = 53,100

NOTES:

P_m = Primary general membrane stress

P_L = Primary local membrane stress

P_b = Primary bending stress

P_e = Secondary expansion stress

Q = Secondary membrane plus bending stress

Fatigue Stress Evaluation (Per NB-3222.4)

Load Pair	Stress Cycles	Stress Amplitude	Cumulative Usage Factor
1. OBE + TH + SRV	50	45,780	0.0061
2. TH + SRV	1,750	43,689	0.1922
3. SRV	7,200	25,258	0.3381

Notes:

OBE = 50 stress cycles total

TH = Thermal transient from SRV discharge, 1800 occurrences

SRV = SRV discharge induced structural vibration, 5 stress cycles x 1800 occurrences.

Emergency Condition Stresses (psi)

	Analyzed	Allowable
$P_L + P_b$	22,652	$1.8\ S_m = 31,860$
Triaxial	25,499	$4.6\ S_m = 81,420$

Local Stress, Thermal Gradient Stress, and Fatigue Factor

A. Localized Bending Stress - The B&PV code has no specific allowable limit for localized bending stress alone. Instead, the code imposes limits on the combinations of primary local membrane stress, the primary bending stress, and the secondary stresses. All those combinations were tabulated previously and shown to be within the limits permitted by the code.

B. Thermal Gradient Stress - The transient heat transfer analysis showed that the maximum linear temperature difference across the pipe wall is 176° F, and the maximum thermal gradient stresses are + 32,530 and -33,908 psi. Those data are included in the upset condition stresses and the fatigue evaluation.

C. Fatigue Factor - The cumulative fatigue life usage factor is 0.3381 for an allowable of 1.0.

CONCLUSIONS

The detailed analyses have shown that NB-3200 code requirements have been met. Therefore, the pipe is qualified stresswise as a Class-1 component. Since Class 3 components are designed to less stringent rules than for Class-1 components, the evaluation performed meets ASME code requirements. Specifically, the structural integrity of the pipe both as a pressure boundary and as a link of the quencher support load path is satisfactory.

REFERENCES

1. NRC letter dated 12/14/82 from Mr. Youngblood to CEI, Mr. Murray R. Edelmann, "Request for additional information on Safety/Relief Piping and Quencher Device"

2. GAI Dwg. No. 04-4549/D-314-011 Sheet 29, Rev. 7, "The Cleveland Electric Illuminating Co., Perry Nuclear Power Plant Unit 1, Main Steam Relief Valve Discharge to Suppression Pool"

3. GAI Dwg. No. 04-4549/S-322-605 Sheet 207.1-207.4 Rev. C, "Pipe Support MK-1B21-H207"

4. GAI Piping Engineering Calculation P-203 Perry Nuclear Power Station "Thermal Hydraulic Transient Force on the Main Steam SRV Discharge Piping" Revision 0.

5. GAI Design Specification DSP-B21-1-4549-00 "Nuclear Boiler System Piping and Pipe Supports ASME III, Division 1" Perry Nuclear Power Plant - Units 1 and 2 Cleveland Electric Illuminating Company Revision 2.

6. NB-3200P/M107A Computer Program "ASME Sec. III NB-3200 Code Procedure to Evaluate Safety Related Class-1 Piping Stress" users manual Vol. 1, 2, 3 Rev. 0.

7. GAI Piping Engineering Calculation P-531 "Perry Nuclear Power Plant - Unit 1. B21 Quencher Supports Revision 1."