
TECHNICAL SERVICES BUSINESS
GE Nuclear Energy
175 Curtner Avenue, San Jose, CA 95125

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INTERNAL CORE SPRAY LINE FLAW EVALUATION AT COOPER NUCLEAR STATION

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Prepared for

Cooper Nuclear Station
Nebraska Public Power District

Prepared by

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Prepared by:
for Rachelle Daniel, Engineer
Engineering & Licensing Consulting Services

Verified by:
H.S. Mehta, Principal Engineer
Engineering & Licensing Consulting Services

Approved by: for D.S. Braden
D.S. Braden, Project Manager
Engineering & Licensing Consulting Services

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1. Purpose/Objective

This analysis documents the results of fracture mechanics evaluation of the indications identified by the in-vessel visual inspection (IVVI) and ultrasonic (UT) inspection of the core spray internal piping during the current refueling outage at the Cooper Nuclear Station. A total of three indications were discovered, two in the A loop and one in the B loop. The first indication was located on the thermal sleeve (Weld #1 , A-Loop) with an estimated length of 8.9 in. The second indication was located on the second thermal sleeve of the same loop (Weld #21) with an estimated length of 5.5 in. The third indication was found on the pipe side near the tee-box (Weld #12, B-Loop) with an estimated length of 1.5 in. Figures 1 and 2 graphically show the locations of these indications.

The analysis consisted of determining the allowable flaw sizes based on the design loadings for the core spray internal piping and a comparison with the projected length of the indications at the end of next fuel cycle considering crack growth.

2. Methods

1. Create an ANSYS (Reference 1) model for the core spray line. Determine the membrane and bending stresses considering various design loadings.
2. Having found the applied stresses at the location of the indications, use the limit load methods of Paragraph IWB-3640, Section XI, ASME Code (see References 2 and 3) to determine the allowable flaw lengths.
3. Determine the indication lengths, including projected crack growth, at the end of next fuel cycle and compare with the allowable values.

3. Assumptions

1. The indications are conservatively assumed to be through-wall even though verified to be part through-wall.
2. Other assumptions are listed throughout the document.

4. Design Inputs

A finite element model consisting of one loop of the internal core spay piping was developed to determine the stresses from various design loads. Figure 3 shows a line plot of the finite element model.

The design inputs in this evaluation consisted of: (1) the geometry of the internal core spray line, (2) the design loads, and (3) the indication dimensions and locations. The geometry of the internal core spray line was obtained from the drawings listed in Reference 4. The design loads considered are the following:

- Internal pressure: During normal operation the pressure differential between the inside and the outside of the line is essentially negligible. The internal pressure during core spray operation is 150 psi. Although a simultaneous occurrence of two upset condition events (i.e., core spray operation and seismic OBE) is judged to be highly unlikely, this internal pressure was used in the evaluation along with the seismic OBE loading. The membrane stress due to internal pressure was calculated using the strength of material formulas.

- Weight: The weight loading including the weight of the contained water was simulated in the ANSYS run by specifying one 'g' acceleration in the vertical direction. The density of the piping material specified in the ANSYS run was a modified value that included the weight of contained water.

- Seismic Inertia (OBE): The seismic analyses of RPV including the internals are documented in References 5 through 7. These analyses provided the base acceleration for an equivalent static analysis of the piping which showed that a '5g' acceleration would conservatively predict the seismic OBE stresses. Therefore, '5g' accelerations in the two horizontal directions (radial and tangential) were applied to the ANSYS model of the core spray piping to conservatively determine the seismic OBE stresses.

- Seismic Anchor Displacement (OBE): Based on the Cooper seismic analyses documented in References 5 through 7 and other more detailed analyses of similar plants conducted by GE, it was conservatively estimated that a 1/4 inch relative seismic anchor motion between the core spray nozzle and the attachment point on the shroud, is possible. Therefore, a 1/4-inch radial displacement was applied at the shroud anchor points (nodes 1 and 79) in the ANSYS run.

- Thermal Anchor Displacement: When the RPV heats up from room temperature to operating temperature, the two anchor points of the internal core spray line (the core spray nozzle and the shroud attachment point) grow vertically and horizontally at different rates due to differences in materials (low alloy steel for nozzle versus stainless steel for shroud). The following displacements were applied at various nodes to account for these effects:

Nodes 1, 79:	displaced 0.444 inches radially due to thermal expansion in the shroud
Nodes 1, 79:	displaced 1.288 inches vertically due to thermal expansion in the shroud
Node 44:	displaced 0.377 inches radially due to thermal expansion in the vessel

- Core Spray Flow Load: This load results when the core spray flow is turned on. The membrane stress due to this load was conservatively calculated as 250 psi.

The direct and bending stresses from each of the preceding loads were first determined either by strength of material calculation or by ANSYS run, and then were summed absolutely to obtain total membrane and bending stresses. The calculated values of the total membrane and bending stresses at the three critical locations in the core spray piping are summarized in the following table:

Stress Summary

Location	Membrane (psi)	Bending (psi)
Thermal Sleeve	1155	1431
Coupling	1029	2492
Tee-Box	1016	1095

5. FRACTURE MECHANICS EVALUATION

5.1. Allowable Flaw Length Determination

The stresses from the table in the preceding section were utilized to determine the acceptable through-wall flaw sizes based on the methods of References 2 and 3. The acceptable flaw size was determined by requiring a safety factor. In the limit load theory, it is assumed that a pipe with a circumferential crack is at the point of incipient failure when the net section at the crack develops a plastic hinge. Plastic flow is assumed to occur at a critical stress level, σ_f, called the flow stress of the material. The flow stress was taken as $3S_m$ (S_m=16.9 ksi for Type 304 stainless steel at 550°F). A safety factor of 2.8 was used as specified in Reference 2 for the normal/upset conditions.

Consider a circumferential crack of length, $l = 2R\alpha$ and constant depth, d. In order to determine the point at which collapse occurs, it is necessary to apply the equations of equilibrium assuming that the cracked section behaves like a hinge. For this condition, the assumed stress state at the cracked section is as shown in Figure 4 where the maximum stress is the flow stress of the material, σ_f. Equilibrium of longitudinal forces and moments about the axis gives the following equations:

(For neutral axis located such that $\alpha + \beta < \pi$)

$$\beta = [(\pi - \alpha d/t) - (P_m/\sigma_f)\pi]/2$$

$$P_b' = (2\sigma_f/\pi)\,(2 \sin\beta - d/t \sin\alpha)$$

where, t = pipe thickness, inches
α = crack half-angle as shown in Figure 4
β = angle that defines the location of the neutral axis
P_m = Membrane axial stress
P_b' = Failure Bending stress

[illegible]y factor is then incorporated as follows:

$$P_b' = SF\,(P_m + P_b) - P_m$$

For the purpose of this evaluation, all three indications were assumed as through-wall. The calculated values of the allowable flaw sizes at the three locations are summarized below:

Indication	Allowable Flaw Length (in)
Weld #1, Loop A	11.8
Weld #21, Loop A	11.8
Weld #12, Loop B	10.7

5.2. Crack Growth Evaluation

Prior crack growth analyses performed for BWR shroud indications have conservatively used a crack growth rate of $5x10^{-5}$ inch/hour.

The stresses induced in the core spray line are very low, as evidenced by the stress results presented in the next section. Those stress results also conservatively include the effects of seismic and core spray injection loads, which are not typically present. Therefore, the applied stress intensity factor is low, and the corresponding crack growth rate would be significantly below the upper bound value of $5x10^{-5}$ inch/hour used here.

Pre-operational testing of BWR internals has demonstrated that high cycle fatigue resulting from flow induced vibration is not a concern for the core spray piping. Additionally, low cycle fatigue caused by assumed thermal transients which could be potentially imposed by cold fluid injections through the feedwater spargers located directly above the core spray lines have been found to be insignificant. Therefore, fatigue crack propagation of indications in the core spray lines is concluded to be negligible, and is not considered to be a further contributor to the crack growth values discussed here.

A crack growth rate of $5x10^{-5}$ in/hr translates into a crack length increase of $(2x5x10^{-5}x12000)$ or 1.2 inches assuming a 18-month long fuel cycle. The factor of 2 in the preceding parenthesis is to account for the growth at each end of the indication.

5.3. Comparison with Allowable Values & Summary

The crack growth values determined in the preceding subsection were added to indication lengths reported in Section 1 to obtain projected indication lengths at the end of next fuel cycle. The following table shows a comparison of these projected indication lengths and the allowable values calculated earlier.

Indication	Current Length (in.)	Crack Growth (in.)	Length at Next Cycle (in.)	Allowable Value (in.)
Weld # 1, Loop A	8.9	1.2	10.1	11.8
Weld # 21, Loop A	5.5	1.2	6.7	11.8
Weld # 12, Loop B	1.5	1.2	2.7	10.7

It is seen that all of the projected indication lengths are less than the corresponding allowable lengths. Based on this it is concluded that the operation in as-is condition of the internal core spray piping is justified for the next fuel cycle.

6. References

[1] DeSalvo, G.J., Ph.D. and Swanson, J. A., Ph.D., ANSYS Engineering Analysis System User's Manual, Revision 4.4, Swanson Analysis Systems, Inc., Houston, PA, May 1, 1989.

[2] ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, American Society of Mechanical Engineers, 1989 Edition, Paragraph IWB 3640.

[3] Ranganath, S. and Mehta, H. S., "Engineering Methods for the Assessment of Ductile Fracture Margin in Nuclear Power Plant Piping," Elastic-Plastic Fracture: Second Symposium, Volume II - Fracture Resistance Curves and Engineering Applications, ASTM STP 803, C.F. Shih and J. P. Gudas, Eds., American Society for Testing and Materials, 1983, pp. II-309 - II-330.

[4] Cooper Shroud Drawings, Drawing # 730E854.

[5] "Seismic Response of RPV and Internals of Cooper Station Plant," GE Design Analysis Unit Report No. RA 145, December 1969.

[6] "Seismic Response of RPV and Internals of Cooper Station with Stiffer Stabilizer Spring," GE Design Analysis Unit Report No. RA 235, May 1970.

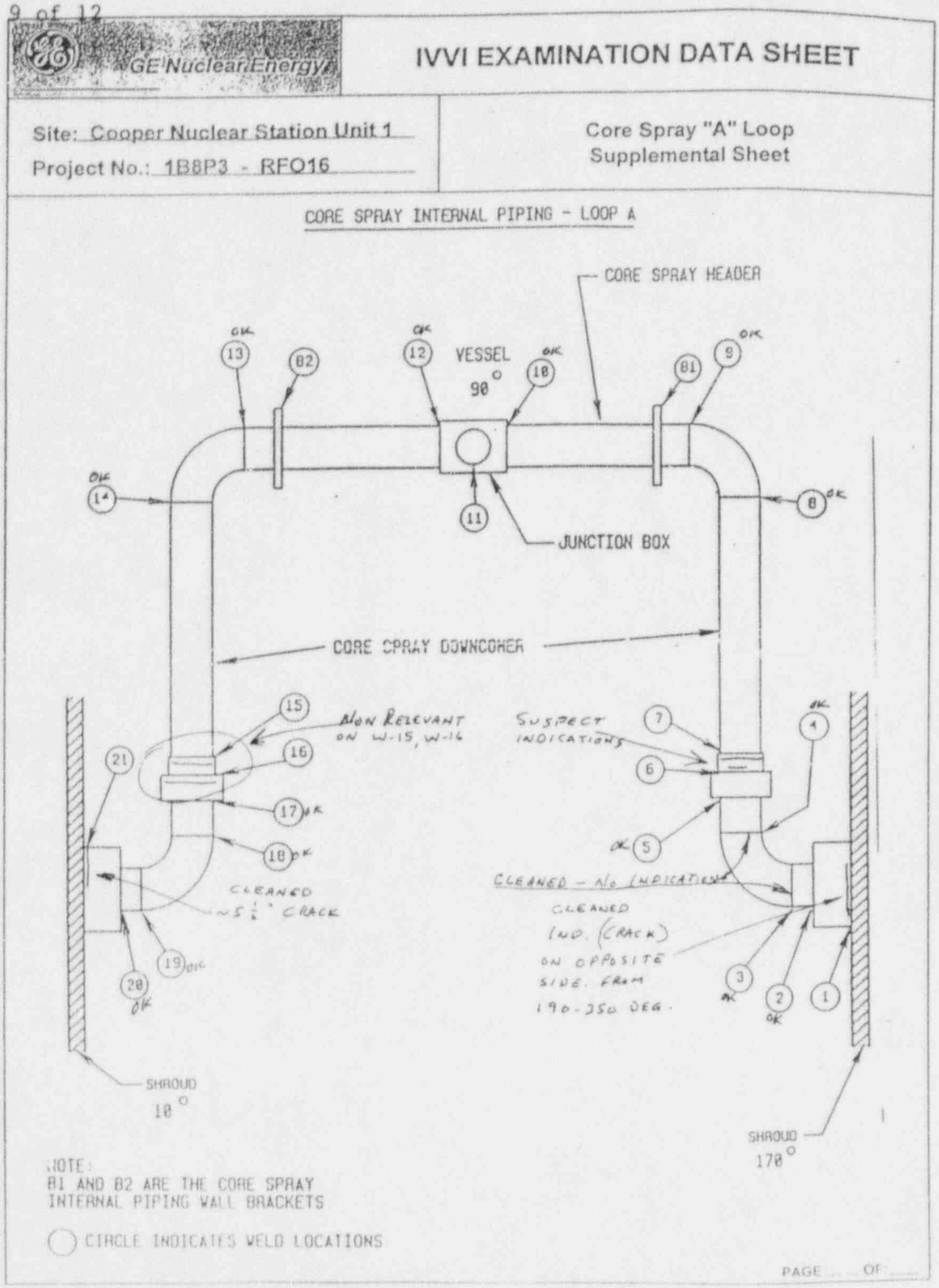
[7] "Structural Analysis Criteria - Appendix C," Cooper Updated Safety Analysis Report (USAR).

7. Units

English units (inches, ksi, ksi$\sqrt{in}$) are used.

8. Conclusions

A flaw evaluation, consisting of stress and fracture mechanics analyses, of the Cooper internal core spray piping was conducted considering the three indications detected during the examinations of current refueling outage. The procedures of Paragraph IWB-3640, ASME Section XI, were used in determining the allowable flaw lengths. The results indicate that the detected indications are projected to be less than the allowable lengths at the end of next fuel cycle. Therefore, the operation of the internal core spray piping at Cooper in the as-is condition is justified at least to the end of next fuel cycle.



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IVVI EXAMINATION DATA SHEET

Site: Cooper Nuclear Station Unit 1

Project No.: 1B8P3 - RFO16

Core Spray "A" Loop
Supplemental Sheet

CORE SPRAY INTERNAL PIPING - LOOP A

NOTE:
B1 AND B2 ARE THE CORE SPRAY INTERNAL PIPING WALL BRACKETS

○ CIRCLE INDICATES WELD LOCATIONS

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Figure 1 Examination Sheet for Loop A

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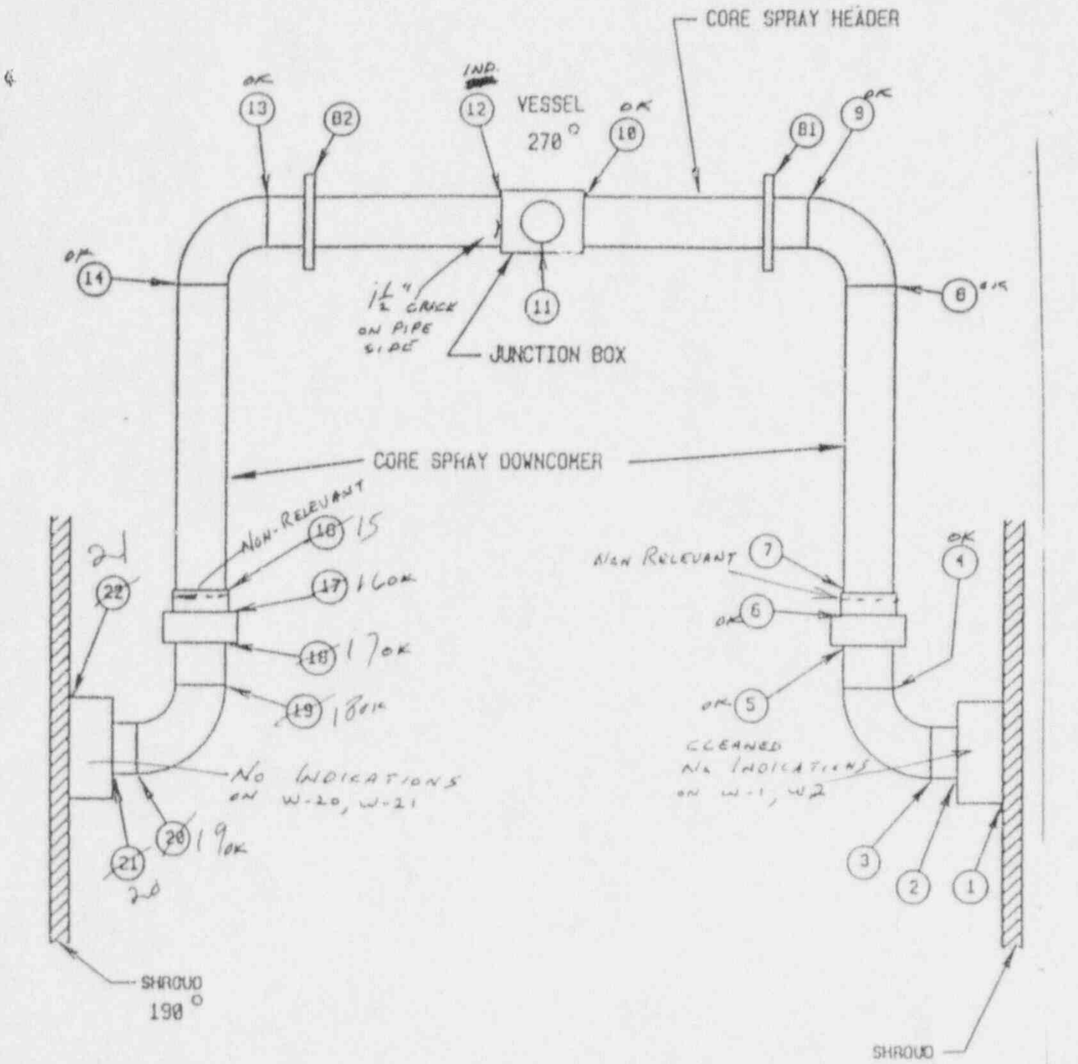
IVVI EXAMINATION DATA SHEET

Site: Cooper Nuclear Station Unit 1

Project No.: 1B8P3 - RFO16

Core Spray "B" Loop
Supplemental Sheet

CORE SPRAY INTERNAL PIPING - LOOP B

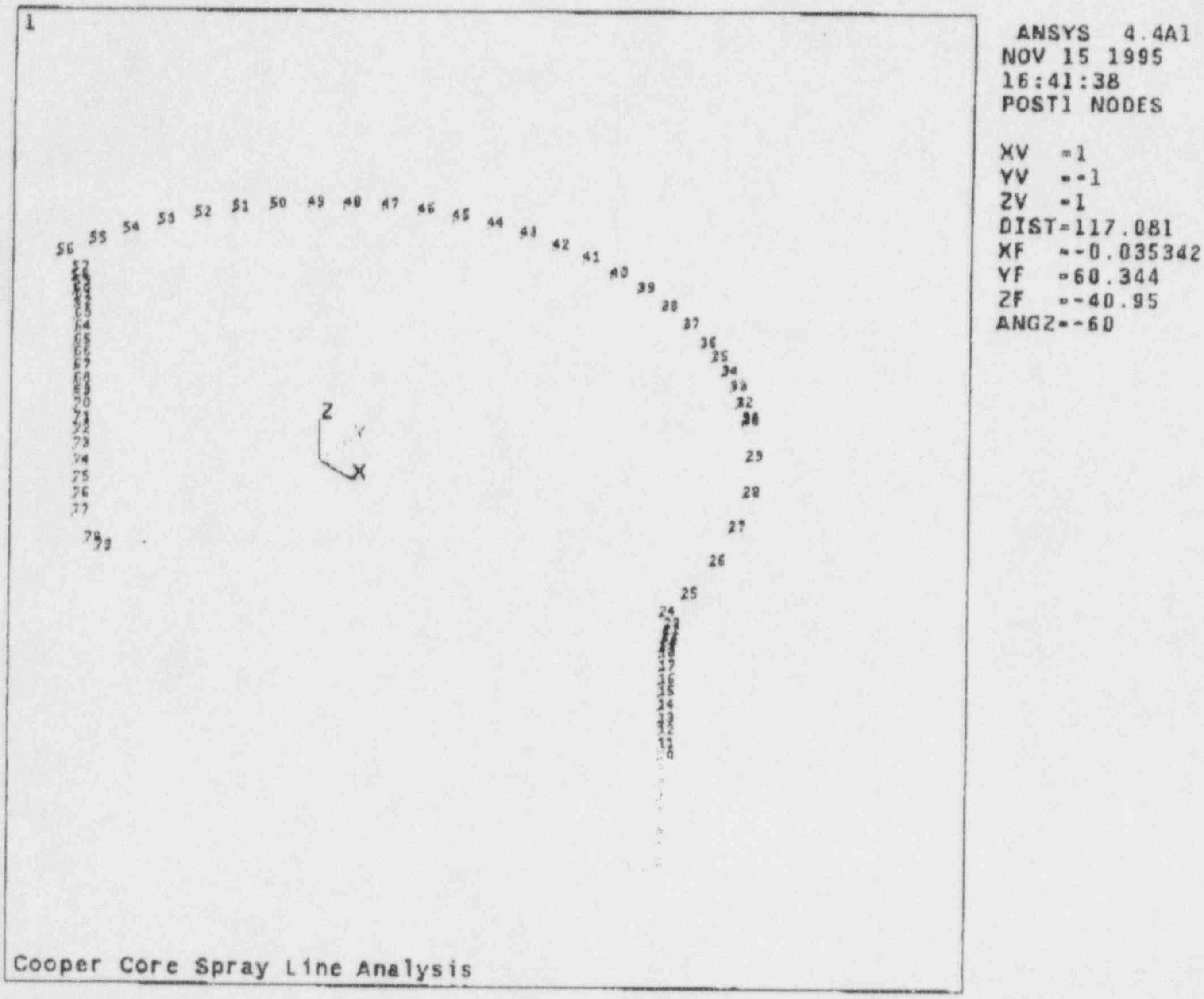


NOTE:
B1 AND B2 ARE THE CORE SPRAY INTERNAL PIPING WALL BRACKETS

○ CIRCLE INDICATES WELD LOCATIONS

PAGE ___ OF ___

Figure 2 Examination Sheet for Loop B



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NOV 15 1995
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ZV   =1
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ZF   =-40.95
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Figure 3   ANSYS Model of the Cooper Core Spray Line

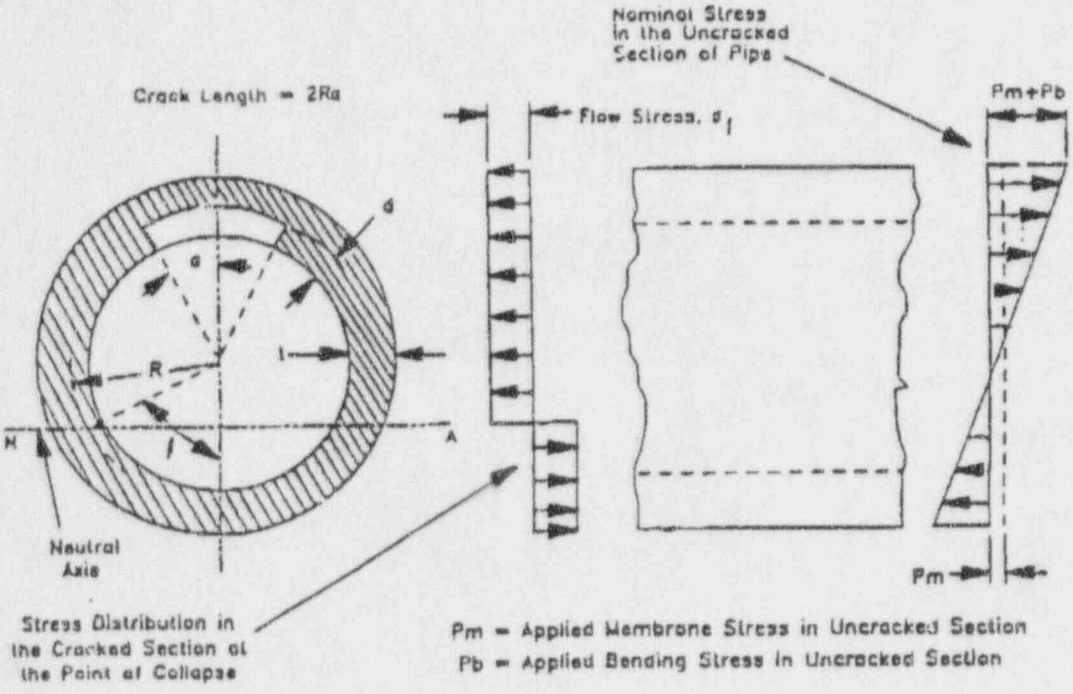

Figure 4 Stress Distribution in a Cracked Pipe at the Point of Collapse

LIST OF NRC COMMITMENTS | ATTACHMENT 3

Correspondence No: NLS950228

The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
The District will continue to inspect the Core Spray Spargers in accordance with IE Bulletin 80-13.	Each refueling outage