**Nebraska Public Power District**
*Nebraska's Energy Leader*

NLS980181
November 06, 1998

Gentlemen:

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Inspection of Reactor Vessel Internal Core Spray Piping
Cooper Nuclear Station, NRC Docket 50-298, DPR-46

References: See Attachment 1

The purpose of this letter is to report the results of the examination of the two flaw indications on the Core Spray piping in the Reactor Pressure Vessel at Cooper Nuclear Station (CNS) performed during the 1998 refueling outage, and to request Nuclear Regulatory Commission (NRC) concurrence that CNS can be safely operated for an additional fuel cycle (Cycle 19). The Nebraska Public Power District (District) requests NRC concurrence prior to startup from Refueling Outage 18 (RFO-18).

Attachment 2 to this letter provides the technical justification for one additional cycle of operation, along with a table stating measured flaw sizes, projected length, and maximum allowable flaw size. In summary, the current size of the indications, and the projected length of the indications at the end of Cycle 19, are well within the maximum flaw length permitted by the 1995 fracture mechanics evaluation (Reference 1), which was approved by the NRC per Reference 3. The District has recently reviewed the 1995 fracture mechanics evaluation, and has determined that the overall conclusions of the evaluation remain valid.

However, as part of the review of the 1995 fracture mechanics evaluation, the District has identified some conservatisms. As such, the District has revised the subject fracture mechanics evaluation to reflect the recent findings. Please find attached (Attachment 3), for NRC review and approval, a copy of revised fracture mechanics evaluation for the subject indications. Approval of this fracture mechanics evaluation is <u>not</u> required prior to startup from RFO-18, but is being transmitted to demonstrate that additional margin exists for crack growth. NRC approval is requested by November 1999. The requested NRC approval would be applicable for Cycles 20 and 21.

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A001



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The District recognizes the importance of reaching a long-term solution regarding the identification of cracks in Core Spray piping. As such, the District is taking a proactive approach on this issue. A list of specific actions regarding the District's approach is discussed in Attachment 2. During the operating cycle, CNS will continue to monitor industry activities in both the repair and replacement areas and the development of improved examination techniques to ensure the long term viability of CNS.

Should you have any questions concerning this matter, please contact me.

Sincerely,

John H. Swailes
Vice President of Nuclear Energy

/kbt/mjf/dnm
Attachment

cc: Regional Administrator
USNRC - Region IV

Senior Project Manager
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector
USNRC

NPG Distribution

# List of References for NLS980181 and Attachments

| Reference Number | Title |
|---|---|
| 1. | Letter (No. NLS950228) to USNRC Document Control Desk from J. H. Mueller dated November 22, 1995, "IE Bulletin 80-13 Response; Visual Inspection of Core Spray Spargers" |
| 2. | Letter (No. NLS950244) to USNRC Document Control Desk from J. H. Mueller dated December 18, 1995, "Follow-up Information to IE Bulletin 80-13 Response" |
| 3. | Letter to G. R. Horn (NPPD) from J. R. Hall (USNRC) dated December 21, 1995, "Cooper Nuclear Station - Evaluation of Core Spray Piping Indications (TAC No. M94097) |
| 4. | Letter (No. NLS960007) to USNRC Document Control Desk from J. H. Mueller dated March 29, 1996, "Impact of Core Spray Line Crack Indications" |
| 5. | Letter (No. NLS960198) to USNRC Document Control Desk from P. D. Graham (NPPD) dated November 27, 1996, "Inspection of Core Spray Spargers and Piping" |
| 6. | Letter (No. NLS970132) to USNRC Document Control Desk from P. D. Graham (NPPD) dated July 14, 1997, "Impact of Core Spray Line Crack Indications" |
| 7. | Letter (No. NLS970088) to USNRC Document Control Desk from P. D. Graham (NPPD) dated May 7, 1997, "Inspection of Core Spray Spargers and Piping" |
| 8. | Letter to G. R. Horn (NPPD) from J. R. Hall (USNRC) dated May 9, 1997, "Cooper Nuclear Station - Evaluation of Core Spray Piping Indications During Refueling Outage 17 (TAC No. M95141)" |

# Justification for One Additional Cycle of Operation

The following discussion provides the basis for why the District has determined that CNS can operate safely for one additional cycle without repair of the subject indications identified in the Core Spray piping.

**Background**

In the Reference 5, the Nebraska Public Power District (District) notified the Nuclear Regulatory Commission (NRC) that Inservice Inspection (ISI) of the Core Spray Spargers and Piping would be performed in accordance with the BWR Vessel and Internal Project Guidelines, BWRVIP-18. In Reference 7, the District reported the results of the inspection of the Core Spray piping and Spargers performed during the 1997 refueling outage. The previously identified indications on the Collar to Shroud weld did not show any significant growth over the cycle and one indication was determined to be non-relevant. Based on these inspection results CNS received permission to operate for one more cycle before reinspecting these indications (Reference 8). In Reference 6, CNS provided an updated evaluation of the impact of cracks in the Core Spray lines.

**Discussion**

The two Loop A Collar to Shroud Welds A-1 and A-21 (also referred to as P8b) discussed in Reference 7 were inspected during the 1998 refueling outage using similar ultrasonic techniques (UT) as employed in the 1997 outage. The flaw indication in Weld A-21 did not exhibit any signs of growth and was measured at 5.6 inches. Weld A-1 exhibited indication growth of 0.4 inches in overall length with a recorded size of 9.5 inches.

Certain enhancements to the UT equipment resulted in greater resolution of the UT display and reduced system noise. Although these changes to the examination system may account for some of the differences in the examination results, the District has conservatively assumed that any increase in the flaw length is due to flaw growth. A summary of the inspection results reported in 1995, 1997, and 1998, and the maximum allowable flaw sizes from the analysis is provided in Table 1.

The District's decision to operate for one additional cycle without repairing the subject indication is based on the following: 1) adequate structural integrity margin, 2) acceptable loose parts evaluation, 3) adequate Core Spray flow, and 4) monitoring industry developments for improved examination techniques and repair technologies.

## 1. Structural Integrity Margin

Weld A-21 has one indication with a length of 5.6 inches ranging from 309 degrees to 50 degrees. As shown in Table 1, the indication in area A-21 has not exhibited any signs of growth.

Three separate indications in area of Weld A-1 were recorded during the UT examination. The indications range from 185 degrees to 355 degrees. For the purpose of evaluation, the three indications are assumed to be connected with an overall length of 9.5 inches. The indications in area A-1, when evaluated as a single indication, show approximately 0.4 inches of growth. The apparent growth rate of 0.4 inches per 18 month cycle, is significantly less than the bounding rate of 1.2 inches/cycle (Ref. BWRVIP-18) assumed in our previous analyses submitted November 22 and December 18, 1995 (References 1 and 2).

A lower than actual system pressure was incorrectly used in the 1995 analysis (References 1 and 2). Correctly applying the actual system pressure decreases the allowable flaw size from 11.8 inches to 11.7 inches. This change is inconsequential and does not affect the overall conclusions of the 1995 fracture mechanics evaluations (References 1 and 2).

The District has reasonable assurance that the allowable flaw size of 11.7 inches provided in our previous submittals will not be exceeded during an additional cycle of operation since: 1) the measured crack growth rate is significantly less than the bounding crack growth rate used in the fracture mechanics analysis, and 2) the flaw size will remain within the allowable size for an additional cycle based on a growth rate of 1.2 inches/cycle.

In order to demonstrate additional structural margin, the District has developed a revised fracture mechanics analysis (Attachment 3) to incorporate the new flaw lengths and CNS specific seismic criteria. The revised fracture mechanics evaluation calculated an allowable flaw size of 13.3 inches. The revised analysis demonstrates that the flaw size will remain within the allowable for as many as three future operating cycles using the bounding flaw growth rate of 1.2 inches/cycle, and as many as nine operating cycles using the measured flaw growth rate of 0.4 inches/cycle.

The revised fracture mechanics analysis was developed to demonstrate that additional margin exists between the known flaw length and the maximum allowable flaw length. NRC approval of this evaluation is not required immediately based on the existing flaw indications being bounded by the existing/previously approved analysis. However, approval of the revised analysis is requested by November 1999, in preparation for the next refueling outage. The requested NRC approval would be applicable to Cycles 20 and 21. NRC approval of the revised fracture mechanics evaluation will permit the District to consider longer-term and more effective corrective actions to address the Core Spray piping issue.

**2) <u>Loose Parts Evaluation</u>**

On March 29, 1996 (Reference 4), the District submitted an evaluation of the potential effects of loose parts resulting from a break in the Core Spray piping at the identified crack locations. This evaluation was reaffirmed in 1997 for the past fuel cycle (Reference 6). The loose parts evaluation remains valid for the next fuel cycle.

**3) Core Spray Flow**

The District has completed a review of the effect of potential leakage through crack indications on peak cladding temperature. The evaluation assumes a through wall crack with a length equal to the maximum predicted flaw size of 13.3 inches and a width of 10 mils. The evaluation also considers the unrelated leakage from the T-box vent hole associated with the Core Spray Loop A piping/core shroud penetration. The calculated maximum leakage rate for Loop A sums to 77.8 gallons per minute (gpm).

CNS has recently updated calculations associated with Core Spray flow to the vessel. The current calculations removed a conservative error in determining the flow to the vessel. The current minimum projected flow, at 113 pounds per square inch differential (psid) reactor pressure, is 5,125 gpm. This is well over the Technical Specification of 4,720 gpm, and shows that the calculated leakage through the T-box and maximum flaw size will have no impact on ensuring the required flow is delivered to the core.

CNS implemented the SAFER/GESTR LOCA analysis concurrent with implementation of Improved Technical Specifications. The CNS analysis was performed utilizing a relaxed Core Spray flow of 4,250 gpm to the vessel at 113 psid reactor pressure. The LOCA analysis assumes a leakage of 100 gpm through the T-Box vent hole, and a resulting flow to the core of 4,150 gpm. Therefore, there is additional margin for leakage beyond that indicated above.

Previous evaluations of flaw leakage were based on the SAFE/REFLOOD LOCA analysis for CNS and the assumption that CNS did not have any excess flow capacity. Therefore a flow reduction would have an impact on the peak clad temperature (PCT). With the excess flow capacity, there is considerable margin with the Core Spray flow and it is not necessary to evaluate PCT or MAPLHGR limits. The bounding MAPLHGR used in the CNS SAFER/GESTR LOCA analysis was 14.0 kW/ft, which is higher than the thermal-mechanical MAPLHGR for the GE8X8NB fuel design. Therefore, the MAPLHGR is not limited by LOCA/ECCS considerations.

**4) Examination And Repair Technology**

The District will continue to examine the two indication sites using Ultrasonic Examination techniques again during RFO-19. The District is evaluating examination of the P9 weld for the Spring 2000 refueling outage (RFO-19). This evaluation involves, in part, monitoring industry progress in the development of an examination technique for the P9 weld through participation in the BWRVIP, EPRI acceptance of the examination techniques, and communication with other utilities. The P9 weld is a hidden weld and has been historically inaccessible for examination using existing examination techniques (see attached sketch). Examination of the P9 weld will be evaluated for the Spring 2000 refueling outage (RFO-19). In addition, the District is evaluating various contingency plans including repair options, implementation of which are dependent on industry experience and future inspection results.

**Conclusion**

The absence of significant growth of the indications, the conservative fracture mechanics evaluation previously submitted and adequate Core Spray flow, demonstrate that at the end of the next fuel cycle the District will not exceed the Code allowable flaw lengths or other design limitations. Based on the previous evaluations, the information provided with this letter, and the criteria of BWRVIP-18, there is sufficient justification for one additional cycle of operation. During the operating cycle, CNS will continue to monitor industry activities in both the repair and replacement areas and the development of improved examination techniques to ensure the long term viability of CNS. NRC concurrence for one additional cycle of operation is requested prior to startup from RFO-18.

**Evaluation of Core Spray Piping Indications**
**Refueling Outage 18**

**Table 1**

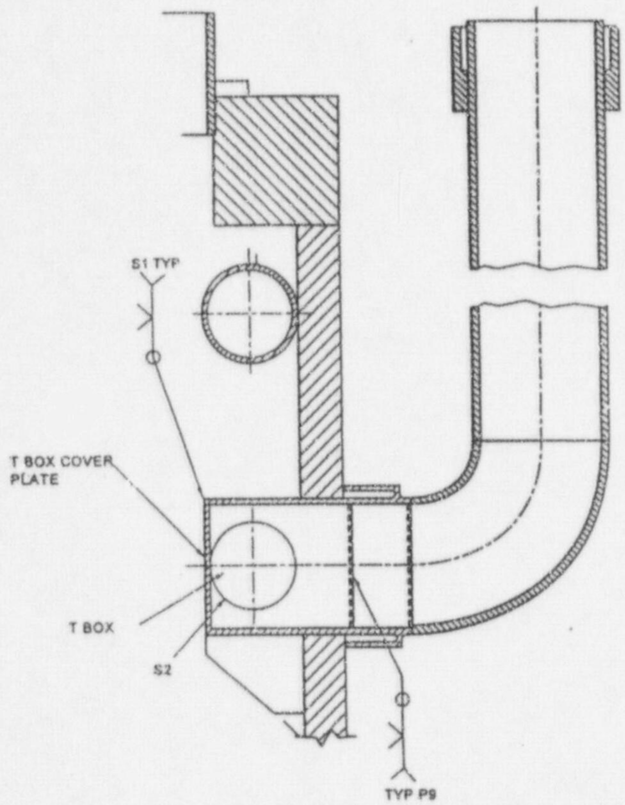
| Weld Number | Length Reported In 1995 (inches) | Length Measured In 1997 (inches) | Length Measured In 1998 (inches) | Projected Length After One Operating Cycle (inches) | Maximum Allowable Flaw Size (inches) |
|---|---|---|---|---|---|
| **A1** | 8.9 | 9.1 | 9.5 | 10.7 | 11.7 |
| **A21** | 5.5 | 5.6 | 5.6 | 6.8 | 11.7 |

**NOTES**

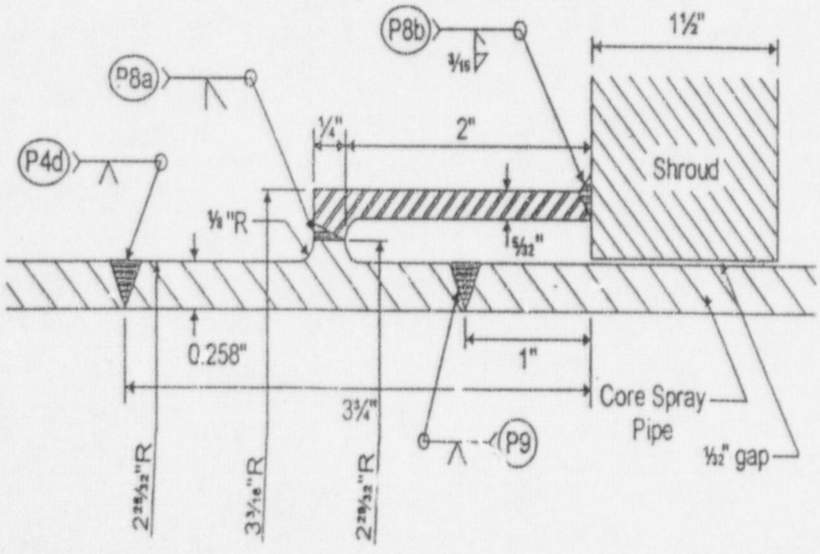
1. Crack growth rate assumed to be 1.2 inches per 18 month operating cycle.

2. The maximum allowable flaw size is based on the fracture mechanics evaluation provided in the District's submittals dated November 22 and December 18, 1995 (References 1 and 2), and reviewed by the NRC in its correspondence dated December 21, 1995 (Reference 3). The allowable flaw size has been reduced by 0.1 inches to incorporate the corrected system pressure. The revised fracture mechanics evaluation (Attachment 3) provides a maximum allowable flaw size of 13.3 inches.

# FIGURE 1




A1 & A21



Attachment 3
to NLS980181
9 Pages Total

# Revised Fracture Mechanics Evaluation for Indications found in Core Spray Piping

Engineering Evaluation No. EE1998-107

**Date:** October 28, 1998

**To:** File

**FOR INTRA-DISTRICT BUSINESS ONLY**

**From:** Ronald L. Yantz / Patrick J. Butler ( MPR, Inc. )

**Subject:** Engineering Evaluation of Conservatism in Core Spray Internal Piping Critical Flaw Size Evaluations

**Engineering Evaluation Number:** EE1998-107

**MWR Number (if applicable):** Not Applicable

**Affected System(s):** Core Spray (CS)

**Affected Document:** None

**Attachments:** None

**References:**

[1] PIR 3-50967;
[2] WesDyne International, Inc. Preliminary Report of Automatic Ultrasonic Examination of CNS Core Spray Piping Welds, dated October 25, 1998;
[3] GE's Report Number GENE-523-A121-1195, "Internal Core Spray Line Flaw Evaluation at CNS", dated November 1995;
[4] NPPD letter to NRC, reference NLS950228, dated November 22, 1995;
[5] NPPD letter to NRC, reference NLS950244, dated December 18, 1995;
[6] GE's Report Number GE-NE-B13-01805-122, Revision 1, March 1998, "CNS Primary Structure Seismic Model Regeneration and Seismic Analysis.";
[7] EPRI Report Number TR-106740 "BWRVIP-18, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines", July 1996;
[8] GE Process Diagram 161F282BC, rev. N05;
[9] NEDC 94-142, rev. 3, "Core Spray Flows With Minimum Flow Valve Open.";
[10] NEDC 87-162, rev. 4, "CNS Frequency Versus Acceleration Response Spectra Curves.";
[11] ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition.

**Description of Evaluation:**

The purpose of this engineering evaluation is to revise existing critical flaw size evaluations for the seal collar-to-shroud welds in loop A of the core spray system internal piping (welds A1 and A21). The conservatism is associated with the use of assumed peak seismic accelerations instead of calculated accelerations from Cooper in-vessel response spectra and inclusion of expansion stress, $P_e$, in flaw evaluations for nonflux welds. The revised critical flaw length will be compared to the projected A1 and A21 weld flaw lengths at the end of the next cycle.

**Requirements/Design Inputs:**

TYPO REV 11/3/98

The Core Spray (CS) System is an essential system located in the Reactor Building. It is a subsystem of the CSCS (Core Standby Cooling Systems). The CS System is designed to maintain continuity of reactor core cooling for the spectrum of loss-of-coolant accidents. It provides cooling spray water to the reactor vessel upon receipt of the reactor low ~~f~~low water level or high drywell pressure actuation signal to mitigate the consequences of a LOCA. The Core Spray System is required to be functional at all modes of operations. The CS system and its components are designed to Class I seismic loadings and must meet all design requirements per the CNS USAR.

**Detailed Evaluation:**

The existing Internal Core Spray Line Flaw Evaluation performed by GE in November 1995 was prepared before CNS requested that GE develop in-vessel seismic response spectra. In this Flaw Evaluation report (GENE-523-A121-1195), the horizontal seismic accelerations were conservatively assumed to be **5.0g** for the OBE and **10.0g** for the SSE. The corresponding vertical seismic accelerations were assumed to be **1.0g** for the OBE and **2.0g** for the SSE. Per reference [6], the in-vessel response spectra for the assumed values can be significantly reduced.

The guidance found in EPRI document TR-106740 was reviewed to determine how the new data could be used to obtain less conservative allowable flaw limits. In appendix B, the EPRI document gives an example of the process used to determine the appropriate horizontal and vertical acceleration to be used in static equivalent analysis from in-vessel response spectra. The approach in Appendix B of the EPRI document, along with in-vessel spectra generated for CNS, will be used to develop less conservative seismic accelerations to be used in static equivalent analysis. Since we know the response at the core shroud attachment points, we can directly use the applicable seismic accelerations at those points. In the example, the fundamental frequency of the piping was conservatively estimated to be 9.6$H_Z$. As discussed in the EPRI document, the CNS fundamental piping frequency would be closer to 25 $H_Z$ because the horizontal pipe sections extending from the nozzle are equal lengths. The appropriate seismic accelerations to use in any new flaw evaluation are determined as follows:

> The horizontal acceleration values are derived from the OBE and SSE response spectra generated for CNS by GE in report GE-NE-B13-01805-122. The accelerations for node point 46 in the EW direction were used to find the listed accelerations. Node point 46 is the elevation 949'-5" core shroud attachment location that is most applicable to the core spray internal piping. The accelerations in the NS direction are slightly lower than the values listed below. A fundamental frequency of approximately 12 $H_Z$ will be conservatively assumed for the internal Core Spray piping.
>
> For an assumed frequency in the 9 to 12 $H_Z$ region, the in-vessel peak spectral accelerations at Node 46 in the EW direction are:
>
> **OBE** = 0.50g
> **SSE** = 0.90g

**Detailed Evaluation:** (continued)

In the static equivalent acceleration method, the peak acceleration is multiplied by a factor of 1.5 to account for the contributions of higher modes.

**OBE** = 0.50g x 1.5 = 0.75g
**SSE** = 0.90g x 1.5 = 1.35g

T use of the multi-mode factor of 1.5 on the horizontal accelerations is maintained for this evaluation, but it too could be reduced. By performing a dynamic response spectra analysis, the effects of the multimode response are captured, and the resulting stresses would be lower than those determined using the static equivalent force / accelerations listed above.

The vertical seismic accelerations applicable to CNS are derived by taking 2/3 of the value derived from the horizontal ground response spectra. The values that would be derived for a frequency of 12 $H_z$ are 0.12g for the OBE and 0.24g for the SSE. These values were determined by using the 854'-9" FRS for the Reactor Building SSE at 0.5% Damping as found in NEDC 87-162 ( ref. 10).

Since the flaw evaluation determined the resultant OBE and SSE stresses, the revised stresses should only be determined by using a common reduction factor on both the horizontal acceleration and the vertical acceleration. Using the equivalent static acceleration method, the revised stresses will be determined by multiplying the stresses from the original evaluation by the ratio of the acceleration. For piping with an assumed fundamental frequency in the 9 to 12 $H_z$ region, following determined reduction factors are applicable:

| | |
|---|---|
| OBE Horizontal new value of 0.75g vs 5.0g, | for a reduction factor of **0.15**. |
| OBE Vertical new value of 0.12g vs 1.0g, | for a reduction factor of **0.12**. |
| SSE Horizontal new value of 1.35g vs 10.0g, | for a reduction factor of **0.14**. |
| SSE Vertical new value of 0.25g vs 2.0g, | for a reduction factor of **0.12**. |

In conclusion, the seismic loading induced stresses found in the original flaw evaluation will be multiplied by a factor of **0.15** for both the OBE and the SSE cases. The reduction in seismically induced stresses should result in a corresponding increase in allowable flaw size that will be determined by using the analysis methods and input stresses found in the attachment to NLS950244. However, a non-conservative value for the assumed internal pressure was found to have been used in the 1995 evaluation.

The original value assumed for internal pressure was 150 psi., as listed in attachment 2 of NLS950228. Under worst-case conditions, the Core Spray System could actually see an internal pressure of 162.7 psi. at the location of the flaw. This value was determined by using the information found on GE process diagram 161F282BC, rev. N05, and resolving the head pressure resulting from the maximum possible Core Spray system flow that was determined in NEDC 94-142, rev. 3.

**Detailed Evaluation:** (continued)

From the process diagram, the Core Spray sparger $\Delta p$ at 6,000 gpm is 320 ft. Resolving the $\Delta p$ for a maximum flow of 6,500 gpm gives the following result:

$$\Delta p_{6500} = \left( (6{,}500)^2 \div (6{,}000)^2 \right) \times \Delta p_{6000} = 1.1736 \times 320 \text{ ft} = \mathbf{375.6 \text{ ft}}.$$

To convert this to psi, multiply by 0.4333, for a final pressure of **162.7 psi**. This means that the analyzed pressure stress should be <u>increased</u> by a factor of 162.7 psi ÷ 150 psi, or **1.085**.

**<u>Stresses and Load Combinations:</u>**

The load combinations used in GE's 1995 flaw evaluation are still applicable in this evaluation. The stresses have been re-calculated based on the Cooper in-vessel seismic response spectra and the increased 1.085 pressure factor. The revised values of the stresses for various operating conditions are summarized below.

**<u>Level A (Normal Operation)</u>**

The revised stresses for the normal condition remain unchanged due to fact that the core spray line does not have any flow or internal pressure during normal operation.

$P_m = 0$ psi

$P_b = 52$ psi

$P_e = 39$ psi

where:
$P_m =$ piping primary membrane stress
$P_b =$ piping primary bending stress
$P_e =$ piping expansion stress (secondary stress including stresses from all displacement-controlled loadings such as thermal expansion, seismic anchor motion, etc.)

**<u>Level B (Upset Condition)</u>**

The revised $P_m$ is 0 + 250 + ( 733 × 1.085 ) + (117 × 0.15 ) = 0 + 250 + **795** + **18** = **1063 psi**.

The revised $P_b$ is 52 + 0 + 0 + ( 1117 × 0.15 ) = 52 + **168** = **220 psi**.

$P_e$ is assumed to be constant, but is probably conservative = **289 psi**.

**<u>Level C (Emergency Condition)</u>**

The revised $P_m$ is 0 + 250 + ( 733 × 1.085 ) + (234 × 0.15 ) = 0 + 250 + **795** + **35** = **1080 psi**.

The revised $P_b$ is 52 + 0 + 0 + ( 2234 × 0.15 ) = 52 + **335** = **387 psi**.

$P_e$ is assumed to be constant, but is probably conservative = **539 psi**.

**Detailed Evaluation:** (continued)

**Level D (Faulted Condition)**

**Case 1**: LOCA Fluid Drag Loads

The revised $\mathbf{P_m}$ is $0 + 0 + 0 + (234 \times 0.15) = 0 + 0 + 0 + \mathbf{35} = \mathbf{35}$ **psi**.

The revised $\mathbf{P_b}$ is $52 + 0 + 0 + (2234 \times 0.15) + 1128 = 52 + \mathbf{335} + 1128 = \mathbf{1515}$ **psi**.

$\mathbf{P_e}$ is assumed to be constant, but is probably conservative = **539 psi**.

**Case 2**: Core Spray Initiation

The revised $\mathbf{P_m}$ **(1080 psi)**, $\mathbf{P_b}$ **(387 psi)**, and $\mathbf{P_e}$ **(539 psi)** stresses for this case 2 of the faulted condition are essentially the same as those for the emergency condition.

**Allowable Flaw Length Calculations:**

The allowable flaw lengths for the two faulted condition cases were calculated using equations (1) and (3) of Appendix C, ASME Section XI [11]. These equations are applicable to through wall flaw configurations. The allowable flaw length in terms of angle $\theta$ was obtained using a circumference of 20.03 inch, corresponding to a sleeve diameter of 6.375 inches. The Appendix C equations are restated below:

$$\beta = [(\pi - \theta \times a/t) - (P_m / 3\, S_m)\, \pi] / 2$$

$$P_b' = (6\, S_m / \pi)(2 \sin \beta - a/t \sin \theta)$$

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

$$P_b = ((P_b' + P_m) / SF) - P_m$$

Where:

$\theta$ = crack half-angle
$\beta$ = angle that defines the location of the neutral axis
$P_m$ = Primary membrane stress
$P_b$ = Primary bending stress
$P_b'$ = failure bending stress
t = pipe wall thickness
a = flaw depth
SF = factor of safety

The flaw depth, **a,** was assumed to be equal to the wall thickness, **t,** for this evaluation. $S_m$ was taken as equal to 16,900 psi. and the SF = 2.77 for this load case. The P8b welds were made by the gas tungsten-arc welding (GTAW) process which is a nonflux welding procedure. As such no "Z" factor will be used in the above equations. As such, Equation 3 of sub-article C-3320 of Section XI, Appendix C is used and the expansion stress, $P_e$, does not factor into the critical flaw sizing evaluations. The original evaluation conservatively added the expansion stress to the primary bending stress, $P_b$.

**Detailed Evaluation:** (continued)

By the same manner used in the NLS document, assume a starting crack half angle, θ, and iterate until $\mathbf{P_b}'$ results in a calculated $\mathbf{P_b}$ equal to the applied $\mathbf{P_b}$.

**Upset Condition**

$P_m$ = 1063 psi.
$P_b$ = 220 psi.
$S_m$ = 16900 psi
SF = 2.77

Assume θ = 2.0924 radians
Then,

β = 0.4917 radians
$\mathbf{P_b}'$ = 2490.5 psi
$\mathbf{P_b}$ = 219.9 psi

The above value of $\mathbf{P_b}$ is close enough to the load value of 220 psi, indicating that the assumed value of θ is correct.

Allowable flaw length = (θ/π) x Circumference
= (2.0924/ π) x 20.03
= 13.3 inches

**Faulted Condition (Case 1)**

$P_m$ = 35 psi.
$P_b$ = 1515 psi.
$S_m$ = 16900 psi
SF = 1.39

Assume θ = 2.3149 radians
Then,

β = 0.4123 radians
$P_b'$ = 2119.6 psi
$P_b$ = 1515.1 psi

The above value of $\mathbf{P_b}$ is close enough to the load value of 1515 psi, indicating that the assumed value of θ is correct.

Allowable flaw length = (θ/ π) x Circumference
= (2.3149/ π) x 20.03
= 14.7 inches

**Detailed Evaluation:** (continued)

**Faulted Condition (Case 2)**

$P_m$ = 1080 psi.
$P_b$ = 387 psi.
$S_m$ = 16900 psi
SF = 1.39

Assume $\theta$ = 2.2286 radians
Then,

$\beta$ = 0.4230 radians
$P_b'$ = 959.38 psi
$P_b$ = 387.18 psi

The above value of $\mathbf{P_b}$ is close enough to the load value of 387 psi, indicating that the assumed value of $\theta$ is correct.

Allowable flaw length = $(\theta/\pi)$ x Circumference
= $(2.2286/\pi)$ x 20.03
= 14.2 inches

Among the two faulted condition cases, the allowable flaw length is the least for case 2. Between the upset condition case and the faulted condition case 2, the upset condition allowable length of 13.3 inches is the least and thus governing.

**RE 18 Inspection Report Data Evaluation:**

The final inspection results show the following flaws on the weld identified as A1:

one flaw indicated from 185deg to 245deg, or 60deg
one flaw indicated from 270deg to 282deg, or 12deg
one flaw indicated from 303deg to 355deg, or 52deg
For a total of **124deg**

However, the three indications are only separated by 25 deg and 21 deg respectively, and therefore the ligament lengths between these flaws are 1.4 inches and 1.2 inches, respectively. Applying the flaw proximity rules of ASME Section XI, Subarticle IWA-3300, a minimum ligament equal to the expected crack growth over the inspected period plus twice the wall thickness is needed to consider two through-wall flaws as separate. Assuming a crack growth rate of 1.2 inch per cycle and given that the wall thickness is 0.156 inches, the minimum required ligament is 1.5 inches. Therefore, the above flaws must be considered to be one continuous flaw of 170deg. The effective flaw length that must be considered is then 9.5 inches.

**Detailed Evaluation: (continued)**

Weld number A1 is the critical P8b weld on the "A" loop of the internal Core Spray piping as identified in the BWRVIP documents. The other P8b weld on this loop of piping is weld number A21, which was found to have an indication from 309deg to 50deg, or 101deg, for a measured flaw of 5.6 inches. By inspection, weld A1 is the critical section with an effective flaw length of 9.5 inches. This value is smaller than the revised allowable flaw length of 13.3 inches, which means the integrity of the critical weld is maintained. The present allowable margin is 13.3 - 9.5, or 3.8 inches. The predicted margin at the end of the next cycle assuming 1.2 inches of crack growth during the cycle is 13.3-10.7, or 2.6 inches

As compared to the 1995 data, in which the effective flaw size on weld A1 was 8.9-inches, the effective flaw length has increased by only 0.6 inches in two operating cycles. At this rate ( 0.3 inch per cycle ), the 3.8 inch margin will be used up in 12 operating cycles. In 1995, the effective flaw length on weld A21 was 5.5 inches, which means that this effective flaw length has not increased significantly in two operating cycles. These actual growth rates are much lower than the bounding predicted growth rate of 1.2 inches per cycle.

Assuming a crack growth rate of 1.2 inch per cycle (18 months), as per the BWRVIP criteria, at least 3 additional operating cycles are expected before the integrity of the weld reaches the analyzed limits. Based on these results, the critical weld should remain acceptable up to at least RE21 unless future inspections show dramatically increased flaw growth.

**Summary of Evaluation and Conclusions:**

The preceding evaluations predict that the critical weld should be acceptable for at least the next two operating cycles. At this time, no repair or replacement of the internal Core Spray piping appears necessary or justifiable. However, if significant new flaws are identified during the inspections scheduled for RE19, the potential need for repair or replacement should be considered.

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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 NL&S Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

| COMMITMENT | COMMITTED DATE OR OUTAGE |
|---|---|
| Re-examine the two indication sites again during RFO-19 | RFO-19 |
| Examination of the P9 weld will be evaluated for the Spring 2000 refueling outage. | RFO-19 |
| CNS will continue to monitor industry activities in both the repair and replacement areas and the development of improved examination techniques to ensure the long term viability of CNS. | During Cycle 19 |
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