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April 8, 1992
C321-92-2055
5000-92-3089

U. S. Nuclear Regulatory Commission
Att: Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219
Revision to NUREG-0619 Routine Inspection
Criteria for Feedwater and Control Rod Drive Return
Line Nozzles

References:

1. GPU Nuclear letter 5000-90-1857, dated January 18, 1990, Feedwater Nozzle Examination.

2. GPU Nuclear letter 5000-90-1954 (C320-90-697), dated July 12, 1990, Feedwater Nozzle and Control Rod Drive Return Nozzle Examinations.

3. GPU Nuclear letter 5000-91-2038 (C321-91-2094), dated April 18, 1991, Control Rod Drive Return Line Nozzle 13R Inspection.

4. NRC letter dated December 13, 1990, GPU Nuclear Proposal to Replace Internal Dye Penetrant Examination of the Feedwater Nozzles and Control Rod Drive Return Nozzle with External Ultrasonic Examinations for Oyster Creek Nuclear Generating Station.

Enclosure: 1. Technical Summary Justifying Nozzle Inspection Method.

NUREG-0619, BWR Feedwater (FW) Nozzle and Control Rod Drive Return Line (CRDRL) Nozzle Cracking, dated November 1980, prescribes periodic Ultrasonic Testing (UT) and Liquid Penetrant Testing (PT) of the OCNGS FW and CRDRL nozzles. It also encourages continued development of UT techniques for nozzle examinations. Improved UT techniques could then form the basis for modifying the NUREG-0619 inspection criteria.

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GPU Nuclear has been evaluating and using the phased-array UT technique, as qualified on mock-ups of FW and CRDRL nozzles, in order to demonstrate sufficient sensitivity to detect flaws that could impact nozzle structural integrity. This technique was successfully used to perform FW and CRDRL nozzle inspections at OCNGS during the 12R and 13R refueling outages, respectively. Previous GPU Nuclear submittals (Reference 1, 2, 3) apprised the NRC of these activities. Additionally, the NRC Staff witnessed a demonstration of the UT equipment on a nozzle mock-up. In response to previous submittals, the NRC concluded (Reference 4) that the UT inspections do provide reasonable assurance of maintaining the structural integrity of the OCNGS FW and CRDRL nozzles.

As a follow-up, based on the successful application of the phased-array UT technique at OCNGS, GPU Nuclear committed (Reference 3) to propose and justify extending the FW and CRDRL NUREG-0619 inspection intervals.

Specifically, GPU Nuclear plans to revise the inspection intervals for these nozzles, as follows:

1. Perform UT inspections of the FW and CRDRL nozzles once each in-service inspection interval in accordance with the ASME Boiler and Pressure Vessel Code Section XI.

2. Eliminate future NUREG-0619 routine PT examinations. Internal PT examinations would only be performed if flaws, which would compromise nozzle integrity, are detected.

3. Reschedule the FW nozzle UT inspection from the 14R refueling outage (scheduled for January 1993) to the 15R refueling outage (scheduled for October 1994).

The basis for the above is provided in the enclosure ("Technical Summary Justifying Nozzle Inspection Method").

The phased-array UT inspection technique has been demonstrated, in mock-up simulations and in-service examinations, to be an appropriate replacement for internal PT examinations in assessing nozzle integrity. In furthering this demonstration, GPU Nuclear has initiated a program to introduce thermal fatigue cracks in both FW and CRDRL nozzle mock-ups to demonstrate the capability of the phased-array UT technique to detect and size such cracks. The results of this test should be available in the latter part of 1992. Additional information regarding this simulation may be found in the enclosure.

During the 12R outage the performance of a UT inspection on the FW nozzle, instead of PT, resulted in estimated savings of 400 Person-Rems. Eliminating future routine PT examinations is expected to produce a similar exposure saving and be commensurate with the ALARA principle.

GPU Nuclear plans to begin implementing the above in-service inspection program during the 15R refueling outage. This coincides with the next scheduled UT inspection of the CRDRL nozzle which was previously inspected during the 13R refueling outage. As such, we have re-scheduled the 14R UT inspection of the FW nozzle so that the program may begin at the same time for all nozzles. Re-scheduling UT inspection of the FW nozzles by one operating cycle to the 15R refueling outage does not pose a safety concern. The FW nozzles were previously inspected in the 12R refueling outage, during the last quarter of 1988. During the examination no indications were found to be reportable. The results of this inspection were submitted to the NRC (Reference 1). Additionally, the enclosed report shows that a presumed crack size equal to the equipment detection sensitivity, subject to thermo-mechanical stresses using design basis operating conditions, requires at least 10 years of operation before exceeding ASME Code Section XI margins for acceptable crack size by analysis. Rescheduling the FW nozzle UT inspection will result in a period of less than six years between inspections of the FW nozzle. Future inspections will be performed at 10-year intervals.

In summary, GPU Nuclear will reschedule the 14R FW nozzle UT inspection to 15R. During the 15R outage, GPU Nuclear will begin implementing UT inspection of the FW and CRDRL nozzles within the in-service inspection program in accordance with ASME Code Section XI. Future PT examinations will also be eliminated unless a flaw, which would compromise nozzle integrity, is detected.

Sincerely,

J. C. DeVine, Jr.
Director, Technical Functions

Enclosure(s)
JCD/EP/plp

cc: Administrator, Region 1
NRC Resident Inspector
Oyster Creek NRC Project Manager

TECHNICAL SUMMARY JUSTIFYING NOZZLE INSPECTION METHOD

PURPOSE

The purpose of this technical summary is to provide a basis for changing the Oyster Creek Nuclear Generating Station NUREG-0619 Ultrasonic Test (UT) inspection intervals and for removing the requirement for liquid Penetrant Test (PT) inspection on a routine basis.

PROPOSAL

Liquid Penetrant Testing (PT), as a primary means to assure structural integrity, has been made obsolete by improved Ultrasonic Testing (UT) inspection coupled with fracture mechanics technology. The UT process qualification accomplished by GPUN includes quantified detection sensitivity and sizing accuracy for surface flaws so that flaws can be dispositioned as acceptable, i.e., not requiring removal, or as unacceptable, meaning that removal is required. Also, periods of operation between UT inspections can now be determined incorporating A.S.M.E., BPVC, Section XI safety margins.

A PT without prior indication of flaws by UT is unnecessary. Internal PT examinations would only be performed if flaws, which would compromise nozzle integrity, are detected.

Also, it is proposed that the UT inspection frequency should be once each Inservice Inspection Interval (ISI) period and not longer than ten (10) years between inspections for both Feedwater (FW) and Control Rod Drive Return (CRDR) nozzles.

BASES FOR PROPOSALS

These proposals are supported by four technical foundations. These are: 1) UT process improvements, 2) design features, 3) performance of each type of nozzle thermal sleeve, and 4) favorable inspection results. Each one of these technical bases is both specific to Oyster Creek and applicable to both FW & CRDR nozzles. Each one of these bases has been previously documented either in submittals or in the open technical literature, except for the second.

UT PROCESS IMPROVEMENTS

GPUN has fabricated full size replicas of both the FW and CRDR nozzles, the latter being clad on the ID surface. Each replica was notched to represent surface flaws of various depths for use as a test block for UT process performance. It was shown statistically (Ref. 1) that the largest flaw in the FW nozzle replica that could escape detection was 0.17?" deep. This is the Oyster Creek UT process detection sensitivity. Also, a correction factor equal to 0.053", as an additive constant, is necessary to account for sizing variation. Similarly for the CRDR nozzle replica, the detection sensitivity was found to be 0.132" and the sizing correction factor was found to be 0.040".

These detection sensitivities support the conclusion that the UT process used at Oyster Creek, the "phased array" method, described in a previous submittal (Ref. 2), can detect flaws that will affect the structural integrity of the nozzle. In the FW nozzle, thermo-mechanical stresses using design basis operating conditions are interacted with a presumed crack size equal to the detection sensitivity in order to show crack propagation (Fig. 1). Similarly, results specific for the CRDR nozzle are shown in Fig. 2. In each instance, at least ten years of operation can occur before exceeding ASME Code, Sect. XI margins for acceptable crack size by analysis.

It is important to note that PT inspection results do not and can not play a part in the process of dispositioning an indication by analysis, as permitted by the ASME Code, Sect. XI, since flaw depth is not indicated by PT.

GPUN has also initiated a program to introduce fatigue cracks in both the FW nozzle replica and the CRDR nozzle replica in order to demonstrate the capability of "phased-array" UT to detect and size actual cracks. We plan to show, as a minimum, that phased-array can detect fatigue flaws that are 0.132" and 0.172" deep. The 0.172" deep flaw will not be clad over since the cladding has been removed from all four feedwater nozzles. The 0.132" deep crack will be implanted as an underclad flaw. That is, the implant will contain a fatigue crack that penetrates 0.132" into the alloy steel then clad over. There will be no flaw penetrating through the cladding. Even though the CRDR cracking did penetrate through the cladding where this phenomena did occur, we consider that detection of underclad cracking is more difficult, because of loss of corner reflector and therefore, more than representative of the ability to detect a through-clad flaw. In addition, thermal fatigue crack growth can be precisely controlled in a single metal. There is currently no technology available for precisely initiating and growing a thermal fatigue crack in a clad, bimetallic assembly.

DESIGN FEATURES OF THE NOZZLE THERMAL SLEEVES

The feedwater nozzle thermal sleeve is the single piston ring type near the inlet with an attached flow baffle near the outlet (Fig. 3). Performance of this design during tests was such that with the spring loaded baffles in contact with the vessel wall ("Gap Sealed" condition in Fig. 4, Ref. 3) increasing leakage during full flow operation does not represent a more severe loading condition because cyclic ΔT_{metal} does not increase. The baffles physically prevent impingement of hot water against the nozzle blend radius. The Oyster Creek specific design is a leak tolerant design, unlike the more common triple thermal sleeve design. The flow baffles will remain in place because of field assembly strategy and local design details. The thermal sleeve piston ring and flow baffles were both seated by hydraulically pushing on the spargers and capturing a "cold spring" load using the pin assembly shown in Sec. B-B of Fig. 4. The cold spring load is about 7,500 lbs. The baffles are screwed into the spargers and crimp locked to the locking ring (Fig. 4 Detail D). The baffles were cold sprung at least 0.140". Finally, the thermal sleeve is rigidly supported by a support ring just upstream of the flow baffles.

The CRDR thermal sleeve performs similarly using a cold-sprung flow baffle attached to an insert that is securely positioned. These are improvements to an already satisfactory original design. The CRDR nozzle has never sustained thermal fatigue cracking. Further, system conditions have been improved by physical modification. The impact of a scram on the CRDR nozzle has been mitigated by maintaining a continuous flow through the nozzle. The original design conditions provided termination of CRD return flow to the vessel during a scram. This cause of nozzle thermal cycling has been eliminated.

PERFORMANCE OF THE NOZZLE THERMAL SLEEVES

Results obtained using the thermal transient monitoring system at Oyster Creek, a modified version of EPRI "Fatigue-Pro" that was described previously (Ref. 2), showed that actual leakage does not exceed 5 gpm during transients. The on-line calculated usage factors for all nozzles during actual plant conditions are insignificant, i.e., <0.001/cycle.

Therefore, both design basis testing and actual performance strongly support on-going operation without causing thermal fatigue cracks.

UT INSPECTION RESULTS

Inspection results of all five nozzles shows no reportable indications from any of the five nozzle inspections used during the phased array UT process. These results have been previously submitted to the NRC (Ref. 2 & 4). The UT inspection results show no reportable indications serving to confirm the conclusion that the nozzle thermal sleeves are adequately designed and perform as designed.

CONCLUSIONS

GPUN has shown that phased array UT is acceptably sensitive for use specifically on the OCNGS reactor vessel feedwater nozzles and CRDR nozzle in place of internal dye penetrant test for at least an interval of ten years. The design features of the thermal sleeve nozzle inserts should provide protection against the initiation of high cycle thermal fatigue cracks and the methods of in-vessel assembly will secure these design features. Results of UT inspections of all nozzles has shown no reportable indications so that the adequacy and integrity of the thermal sleeves is confirmed.

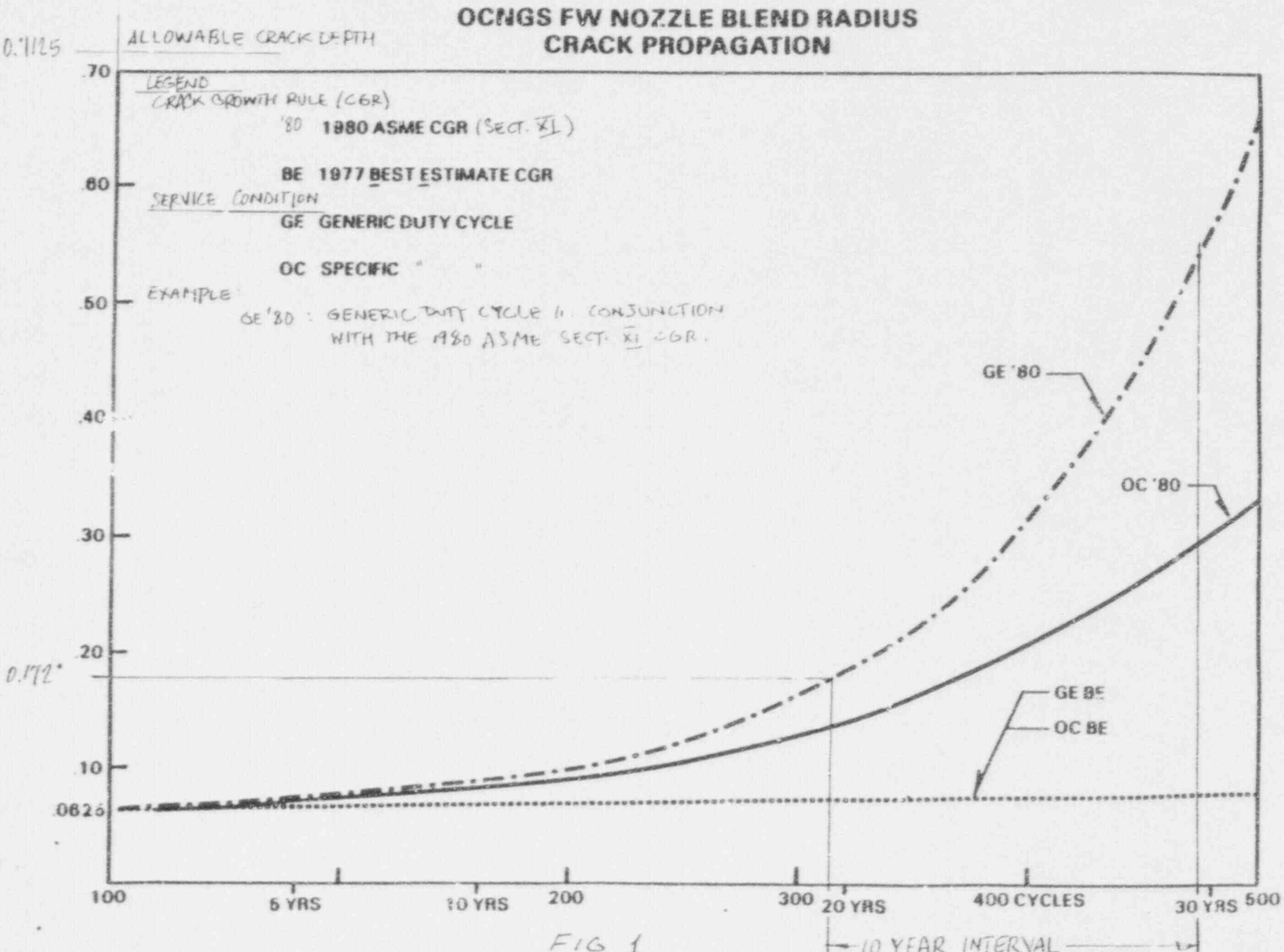
REFERENCES

1. Leshnoff, S. D., and Orski, M. A., Disposition of Feedwater Nozzle UT Indications in a BWR, Proceeding of the American Nuclear Society, June, 1989.

2. J. C. DeVine, Jr., to NRC, Feedwater Nozzle & Control Rod Drive Return Nozzle Examinations, 5000-90-1954, C320-90-697, dated 7/12/90.

3. BWR Feedwater Nozzle/Sparger Final Report (Supplement 2), General Electric Report NEDE-21821-02, August, 1979.

4. Report to NRC providing on CRDR inspection results from 13R. GPU Nuclear letter 5000-91-2038 (C321-91-2094), Control Rod Drive Return Line Nozzle 13R Inspection, dated April 18, 1991.



OCNGS FW NOZZLE BLEND RADIUS
CRACK PROPAGATION
0.7125
ALLOWABLE CRACK DEPTH
LEGEND
CRACK GROWTH RULE (CGR)
'80 1980 ASME CGR (SECT. XI)
BE 1977 BEST ESTIMATE CGR
SERVICE CONDITION
GE GENERIC DUTY CYCLE
OC SPECIFIC " "
EXAMPLE
GE'80 : GENERIC DUTY CYCLE IN CONJUNCTION WITH THE 1980 ASME SECT. XI CGR.
.70
.60
.50
.40
.30
.20
.10
.0625
0.172"
GE '80
OC '80
GE BE
OC BE
100
5 YRS
10 YRS
200
300
20 YRS
400 CYCLES
30 YRS
500
10 YEAR INTERVAL

FIG 1

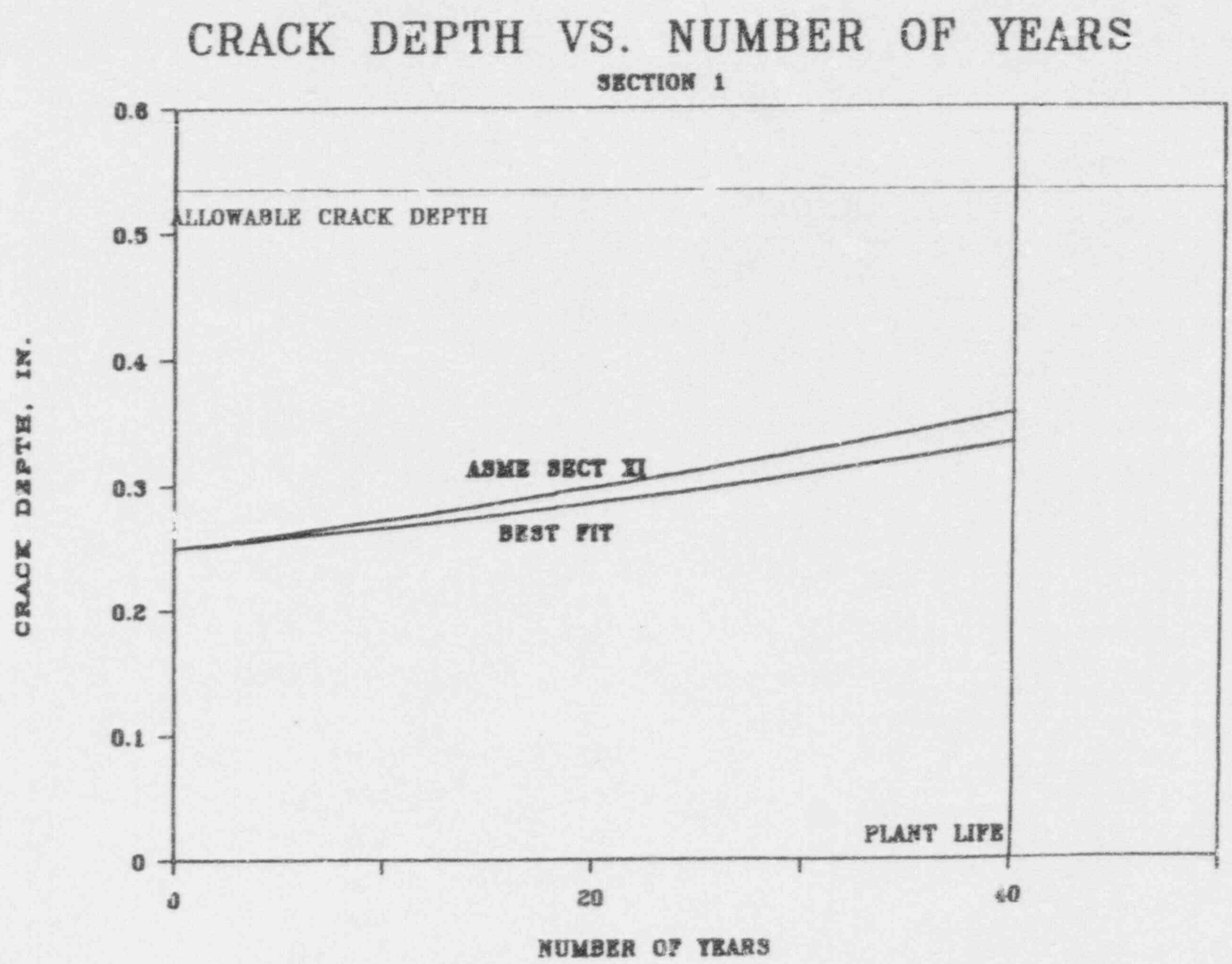


FIGURE 2 Crack Depth Versus Number of Years

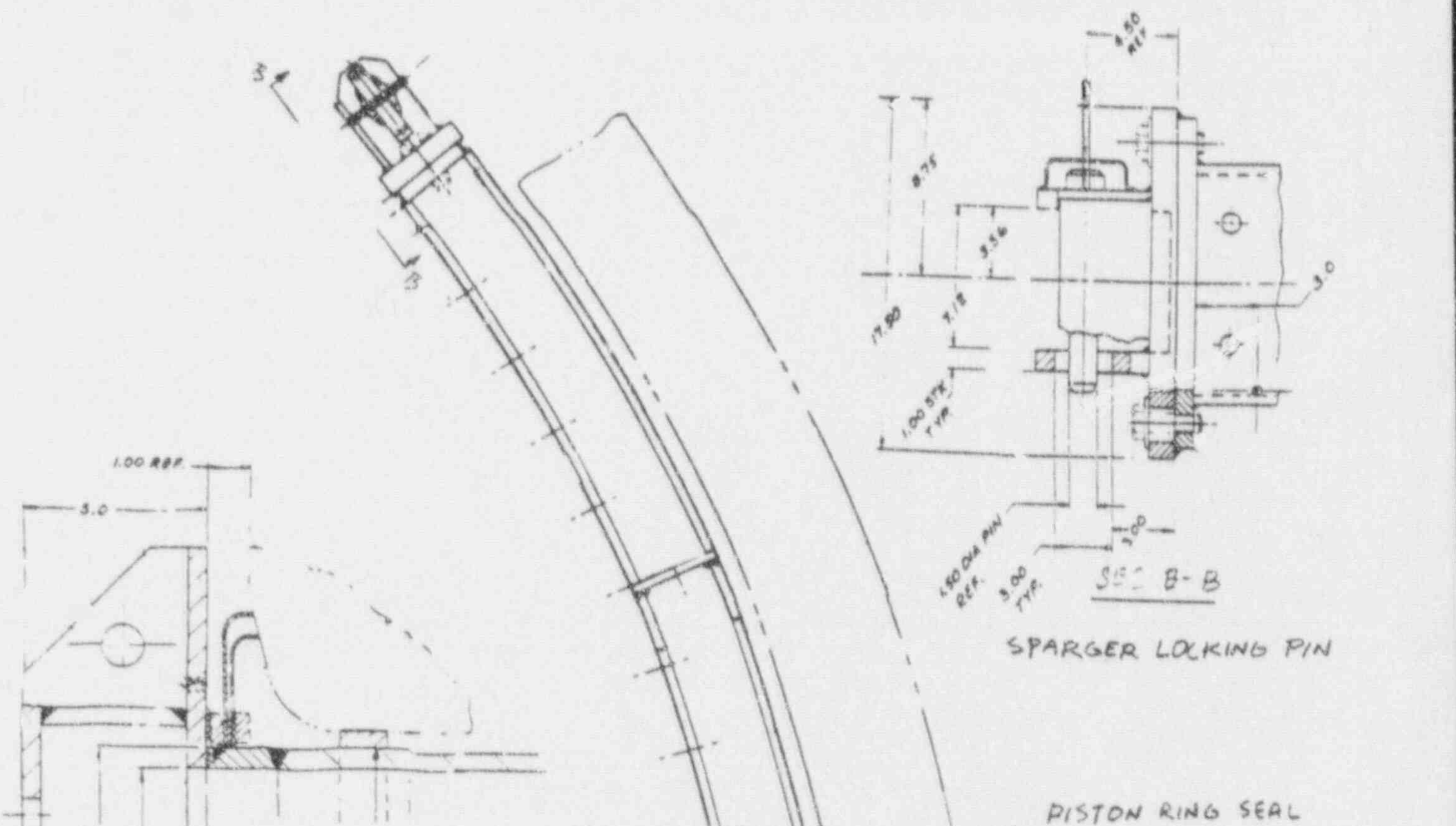


FIG. 3 FEEDWATER NOZZLE THERMAL SLEEVE AND FEEDWATER SPARGER ASSEMBLY

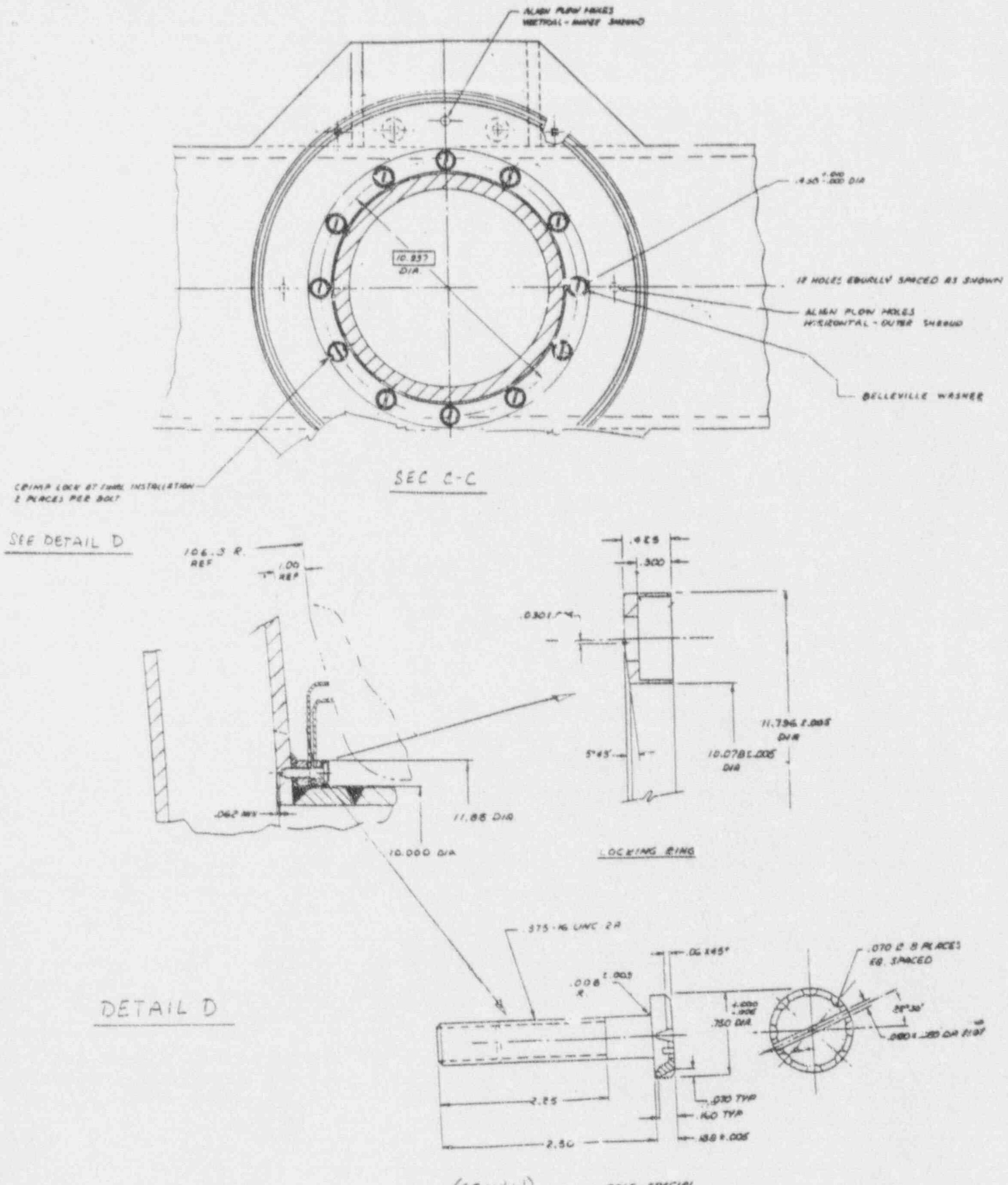


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FIG. 3 ^ FEEDWATER NOZZLE THERMAL SLEEVE AND FEEDWATER SPARGER ASSEMBLY DETAILS.

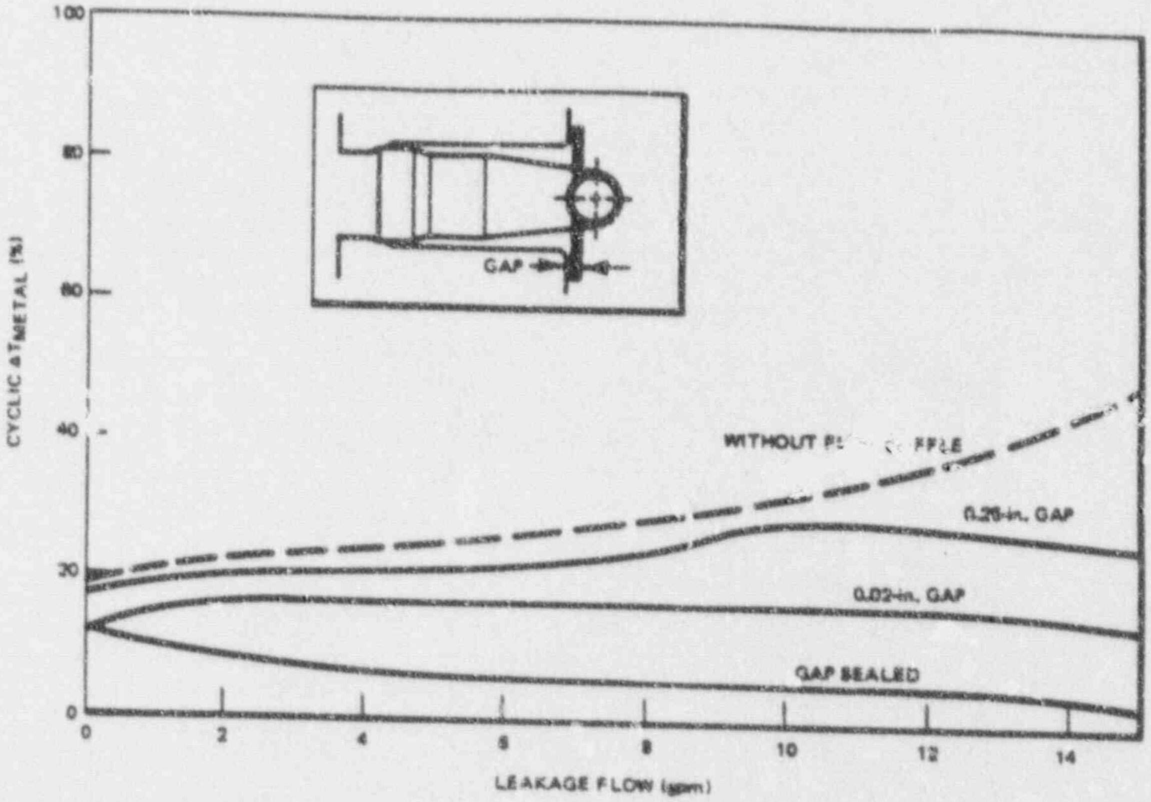

Figure 4 Metal Temperature Cycling at Nozzle Blend Radius with Flow Baffle