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The Northeast Utilities System

OCT 22 1999

Docket No. 50-336
B17868

RE: 10 CFR 50.55a(a)(3)(ii)

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

Millstone Nuclear Power Station, Unit No. 2
Request to Use an Alternative to ASME Code Section XI
Code Case N-619

Pursuant to the provisions of 10 CFR 50.55a(a)(3)(ii), Northeast Nuclear Energy Company (NNECO) hereby requests permission to use an alternative to the ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition. Specifically, NNECO requests to utilize Code Case N-619 "Alternative Requirements for Nozzle Inner Radius Inspections for Class 1 Pressurizer and Steam Generator Nozzles Section XI, Division 1," for the Millstone Unit No. 2 Third 10-Year Interval Inservice Inspection (ISI) Program Plan as detailed in Relief Request RR-89-23 (1).

Code Case N-619 was approved by the ASME Boiler and Pressure Vessel Code Committee on February 15, 1999. Concurrently, the ASME Boiler and Pressure Vessel Code Committee also incorporated this Code Case into the 1999 Addenda of Section XI. This ASME Code Case and the approved ASME Code change eliminated the requirements for nozzle inner radius examinations for the Pressurizer and Steam Generators listed in Table IWB-2500-1, Examination Category B-D, "Full Penetration Welded Nozzles In Vessels - Inspection Program A or B." This concurrent action to change the ASME Code is not the standard practice within the ASME, but does happen in those unique situations where the overall consensus process has determined that to continue to perform these examinations is a hardship that is not conducive to ALARA and has been shown to require an unnecessary burden on Licensees with a negligible safety benefit by the ASME Boiler and Pressure Vessel Code Committee. This Code Case and its related 1999 Addenda of Section XI is not currently included in the NRC approved Code Cases identified in Revision 12 of USNRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability - ASME Section XI Division 1," dated May 1999 or within 10 CFR 50.55a(b)(2).

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Other considerations were explored by the ASME Boiler and Pressure Vessel Code Committee in reducing the burden caused by these examinations. This exploration included reviewing NRC approved relief requests for other Licensees and the alternatives approved in lieu of performing these nozzle inner radius examinations. The review of NRC approved relief requests included a Relief Request that had been submitted and approved in February 1992 for the Haddam Neck Plant (RR 3-23). This Relief Request allowed a VT-1 examination of the clad inner radius sections for the Steam Generator nozzles in lieu of performing the required volumetric examination due to the difficulties associated with ultrasonic examination of the cast head material of the Steam Generators. The Haddam Neck Relief Request was typical of several other Licensees Relief Requests which had been approved by the NRC to use a VT-1 examination.

Use of VT-1 examination in lieu of performing volumetric nozzle inner radius examinations is now considered of very little value based on conclusions cited in a paper written by Dr. F. A. Simonen of the Pacific Northwest Laboratory. This paper, "Clad Failure Models for Underclad Flaws in Reactor Pressure Vessels," published in the Piping and Pressure Vessel PVP-Vol. 280 (June 1994), identified that the cladding on Pressure Vessel nozzles is generally not brittle and that tests conducted at Oak Ridge National Laboratory had shown that underclad cracks could propagate into the vessel wall without any fracture of the cladding material. These points are relevant to the position that use of a VT-1 examination is not an acceptable alternative examination methodology in this situation.

This information was discussed within the ASME Code committees during the development of Code Case N-619 (Attachment 2) and it was determined that, relative to this Code Case, a VT-1 examination was not an acceptable alternative in lieu of a volumetric examination on a clad nozzle inner radius section since such an examination would probably not identify the flaws of concern, if they existed. The document used by the Code committee as a basis to support approval of Code Case N-619 is provided in Attachment 3 and is entitled: "Technical Basis for Elimination of Nozzle Inner Radius Inspections (November 1997)." This "White Paper" provides the key technical points associated with ranges of radiation exposures resulting from examinations, inspection history, descriptions of examination volumes and examination approaches, deterministic fracture evaluations, and risk assessments for the nozzle inner radius sections of concern. These key technical points show that the burden imposed by performing inner radius examinations on the nozzles are unnecessary. The information contained within this "White Paper" is commensurate with the applicable attributes of the Millstone Unit No. 2 Pressurizer and Steam Generator nozzles.

Based on the forgoing and the estimated exposures identified in the attached Relief Request (RR-89-23) (Attachment 1), NNECO plans during the Third 10-Year Interval to eliminate the nozzle inner radius examinations as allowed by Code Case N-619, subject to NRC approval. This determination was based on the hardship that is created as a result of the current requirement to perform these examinations. NNECO also plans to use Code Case N-619 until such time this Code Case is incorporated into a future revision of Regulatory Guide 1.147. At such time NNECO will follow all provisions in Code Case N-619, including any exceptions or limitations as might be provided within the Regulatory Guide.

Review of this request is needed by December 2000.

There are no regulatory commitments contained within this letter.

Should you have any questions regarding this matter, please contact Mr. D. W. Dodson at (860) 447-1791, extension 2346.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: Raymond P. Necci
Vice President - Nuclear Oversight and Regulatory Affairs

BY: ______________________
David A. Smith
Manager - Regulatory Affairs

Attachments: (3)

1. Relief Request RR-89-23
2. ASME Code Case N-619
3. ASME "White Paper" on Code Case N-619

cc: H. J. Miller, Region I Administrator
R. B. Eaton, NRC Senior Project Manager, Millstone Unit No. 2
D. P. Beaulieu, Senior Resident Inspector, Millstone Unit No. 2

Docket No. 50-336
B17868

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Relief From In-Service Inspection Requirements
Request To Use Code Case N-619 As An Alternative To ASME Section XI
Relief Request RR-89-23

October 1999

Attachment 1

Request To Use Code Case N-619 As An Alternative To ASME Section XI

Relief Request: RR-89-23

Code Class: 1 **Zone:** 1-03, 1-04, & 1-15

Code Category: B-D, Full Penetration Welds of Nozzles in Vessels - Inspection Program B

Item No.: B3.120 and B3.140

Code Requirement:

Section XI of the ASME B&PV Code, 1989 Edition, Table IWB-2500-1, Examination Category B-D, requires that Nozzle Inside Radius Sections for the nozzles of the Pressurizer and Steam Generator (Primary Side) be volumetrically examined at least once each inspection interval.

Code Relief Requested:

Pursuant to the provisions of 10 CFR 50.55a(a)(3)(ii), relief is requested to utilize the alternative requirements of Code Case N-619 "Alternative Requirements for Nozzle Inner Radius Inspections for Class 1 Pressurizer and Steam Generator Nozzles Section XI, Division 1," for the Millstone Unit No. 2 Third 10-Year Interval ISI Program Plan. Volumetric examinations for the following Nozzle Inside Radius Sections will be eliminated with this request:

TABLE 1, INSIDE RADIUS SECTIONS			
Nozzles	Component ID	Exam Results[1]	Est. Exposure. Per Exam[2]
Surge Nozzle Bottom. Dead Center	PR-B-IR-1	Satisfactory	750 mRem
Safety Nozzle @ 260 AZ	PR-T-IR-2	Satisfactory	600 mRem

TABLE 1, INSIDE RADIUS SECTIONS			
Nozzles	Component ID	Exam Results[1]	Est. Exposure. Per Exam[2]
Safety Nozzle @ 180 AZ	PR-T-IR-3	Satisfactory	600 mRem
Relief Nozzle Top Dead Center	PR-T-IR-1	Satisfactory	750 mRem
Spray Nozzle @ 315 AZ	PR-T-IR-5	Satisfactory	600 mRem
SG-1 Inlet	SG-1-IR-4-A	Satisfactory	750 mRem
SG-1 Outlet	SG-1-IR-2-A	Satisfactory	750 mRem
SG-1 Outlet	SG-1-IR-5-A	Satisfactory	750 mRem
SG-2 Inlet	SG-2-IR-4-A	Satisfactory	750 mRem
SG-2 Outlet	SG-2-IR-4-A	Satisfactory	750 mRem
SG-2 Outlet	SG-2-IR-4-A	Satisfactory	750 mRem

Notes:
1. Satisfactory indicates examined with no recordable indications.
2. Includes all radiation exposure (e.g., scaffolding, insulation removal & replacement, surface preparation, and ultrasonic examination).

Reason for Relief:

Code Case N-619 has eliminated the requirements for nozzle inner radius examinations for the Pressurizer and Steam Generators listed in Table IWB-2500-1, Examination Category B-D, "Full Penetration Welded Nozzles In Vessels - Inspection Program A or B" The ASME consensus process has approved this Code Case, with NRC member participation, and has determined through this approval process that to continue to perform these examinations is a hardship that is not conducive to ALARA and has been shown to require an unnecessary burden on Licensees for a negligible safety benefit.

The statement above is paraphrased from the conclusion in the supporting ASME "White Paper" that was used for the Code Case and is written as follows: "The results shown in this report have demonstrated that it is highly unlikely that the nozzles considered in this report would fail under any anticipated service conditions. Inservice inspections can hardly benefit plant safety for something that is very unlikely to happen. The inspection is very difficult to perform because of access, and high radiation environment in many cases. Inspections which have been done have not led to discovery of any indications at all. It is recommended that Inservice inspections on all PWR Pressurizer and Steam Generator nozzle inner radius regions be eliminated for economic and health reasons without any risk to structural integrity." The Key technical elements of this "White Paper" are:

- Radiation Exposures To Perform These Examinations (100 MR/Hr To 1 R/Hr Per Nozzle)
- After 25 Years Of Industry Operation No Cracking Incidents Of Any Kind In The Steam Generator Or Pressurizer Nozzle Inner Radius Regions Have Occurred Or Been Identified
- Descriptions Of Examination Volumes And Examination Approaches (Difficult Examination Best Effort In Many Cases)
- Deterministic Fracture Evaluations (Shows Very Small Flaw Growth Over The Entire Operating Life Of the Component)
- Risk Assessment (Credits Construction NDE, Very Little Change In Failure Probability With Or With Out ISI)

In conclusion, as the current examination requirements apply to Millstone Unit No. 2, the total estimated radiation exposure to perform these examinations of 7.8 Person Rem per 10-year interval is excessive. When this radiation exposure is coupled with the key technical elements addressed in the "White Paper" there appears no reason to technically continue to perform these examinations. NNECO has determined from all this information that a firm basis exists to apply for this relief request based on the hardship that will occur if Millstone Unit No. 2 is required to continue to perform these examinations.

Proposed Alternative:

During the Third 10-Year Interval, NNECO plans to use Code Case N-619, subject to NRC approval, until such time this Code Case is incorporated into a future revision of Regulatory Guide 1.147. Upon issuance of the Regulatory Guide, NNECO will follow all provisions in Code Case N-619, including any exceptions or limitations as would be discussed in the Regulatory Guide. Additionally, Code required system pressure tests with VT-2 visual examinations will continue to be performed on the Steam Generators and Pressurizer in accordance with the Millstone Unit No. 2 ISI Program Plan.

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Attachment 2

Millstone Nuclear Power Station, Unit No. 2

ASME Code Case N-619

October 1999

Attachment 2

ASME Code Case N-619

CASE
N-619

CASES OF ASME BOILER AND PRESSURE VESSEL CODE

Approval Date: February 15, 1999

See Numeric Index for expiration and any reaffirmation dates.

Case N-619
Alternative Requirements for Nozzle Inner Radius Inspections for Class 1 Pressurizer and Steam Generator Nozzles
Section XI, Division 1

Inquiry: What alternative to the inspection requirements of Table IWB-2500-1, Examination Category B-D, for pressurizers and steam generators may be used?

Reply: It is the opinion of the Committee that the inspections required by Table IWB-2500-1, Examination Category B-D, Item Numbers B3.40 and B3.60 (Inspection Program A) and Item Numbers B3.120 and B3.140 (Inspection Program B) need not be performed.

Attachment 3

Millstone Nuclear Power Station, Unit No. 2

ASME Code "White Paper"
Technical Basis for Elimination of Nozzle Inner Radius Inspections
(For Vessels Other Than The Reactor Vessel)

October 1999

Attachment 3

ASME Code "White Paper"
Technical Basis for Elimination of Nozzle Inner Radius Inspections
(For Vessels Other Than The Reactor Vessel)

Technical Basis for Elimination of Nozzle Inner Radius Inspections

(For Vessels Other Than The Reactor Vessel)

November 1997

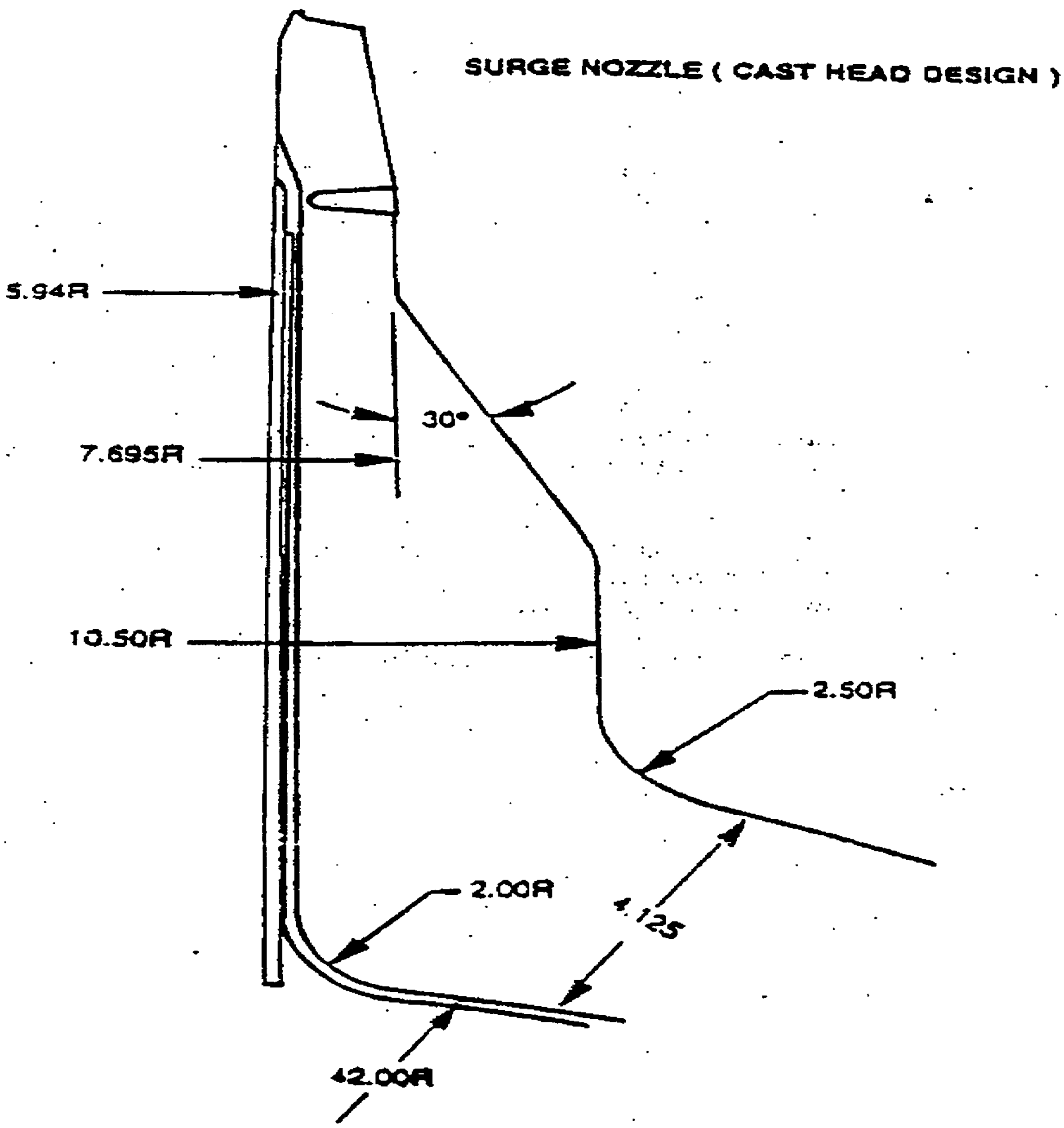
1. Introduction

The requirement for inspection of nozzle inner radius regions in Class 1 systems has been in effect for a very long time, and has not resulted in any inspection findings in any of the vessels and nozzles of interest here, mainly the steam generators and pressurizers of PWRs. The original requirement was included as a result of a cracking event in a non-nuclear vessel which occurred near the time when the ASME Section XI inspection requirements were being established.

The original requirement, as instituted in the early 1970s, was a good idea, since there was only limited experience in operating nuclear plants. Today, after some 25 years of operation, no cracking incidents of any kind in these nozzle inner radius regions have been found whatsoever. It is advisable ,therefore, to eliminate this requirement since it is no longer necessary.

This report provides the technical bases for elimination of this requirement, from both the deterministic and probabilistic view points. First we will describe the extensive inspections performed on the nozzle inner radius regions during the fabrication process, and summarize in-service inspection results obtained over the past 25 years. This will show that there is no evidence of any cracking has ever been found in this region. Second, a series of structural integrity evaluations will be presented covering the range of nozzle geometries of interest here, to demonstrate that these nozzles have a large tolerance for flaws. Third, we will review the general practices currently used by the nuclear industry, along with the results of inspections done on the Westinghouse, Babcock and Wilcox, and Combustion Engineering plants. Risk based evaluations will be performed to demonstrate that failure probability is extremely low under the plant operating conditions and show that there is no change in the risk if the inspections are eliminated.

The range of geometries of the nozzles of interest is shown in Figures 1-1 through 1-6.


SURGE NOZZLE (CAST HEAD DESIGN)
5.94R
7.695R
30°
10.50R
2.50R
2.00R
4.125
42.00R

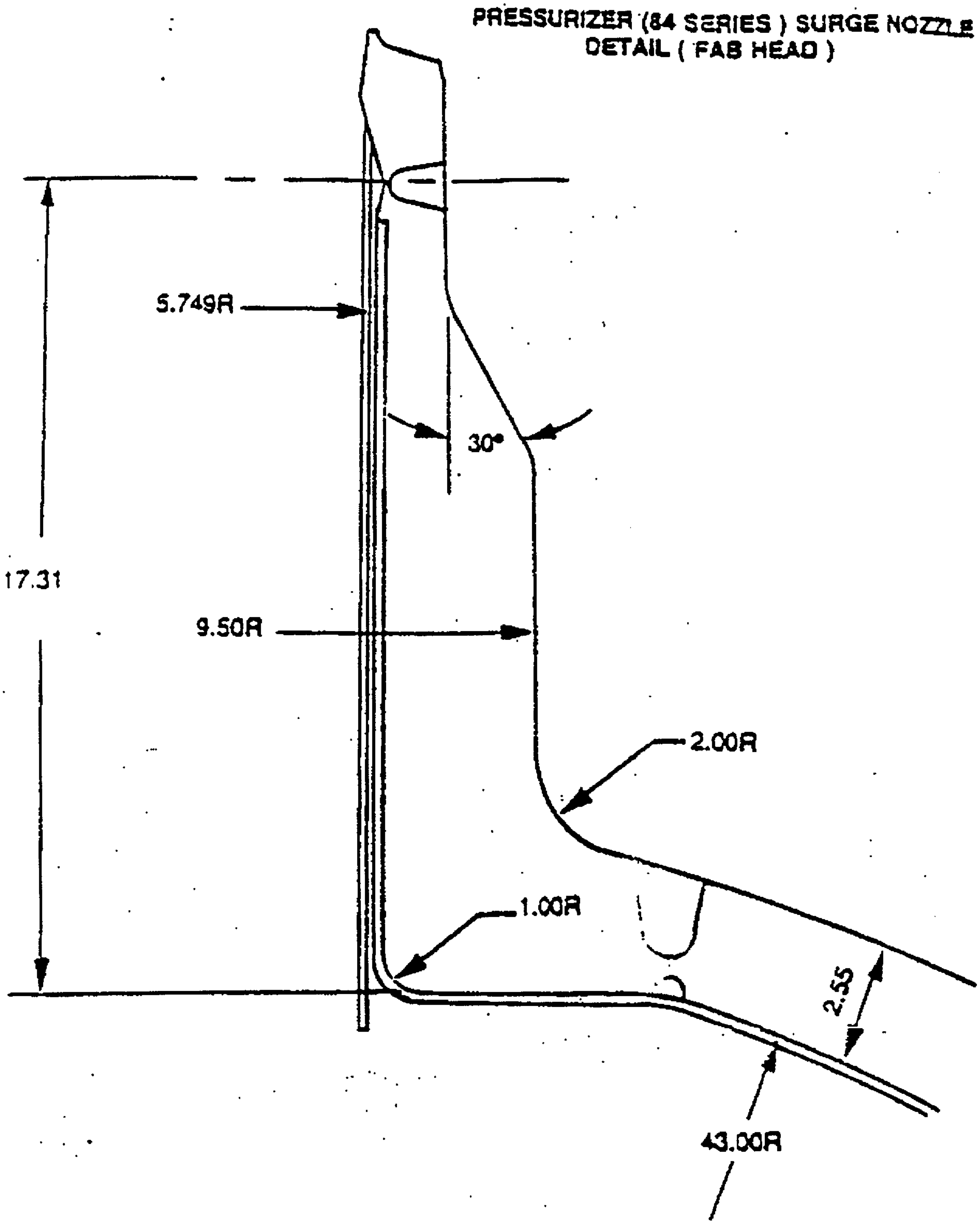


Figure 1-2 Geometry of a Typical Pressurizer Surge Nozzle - Fabricated Head Design

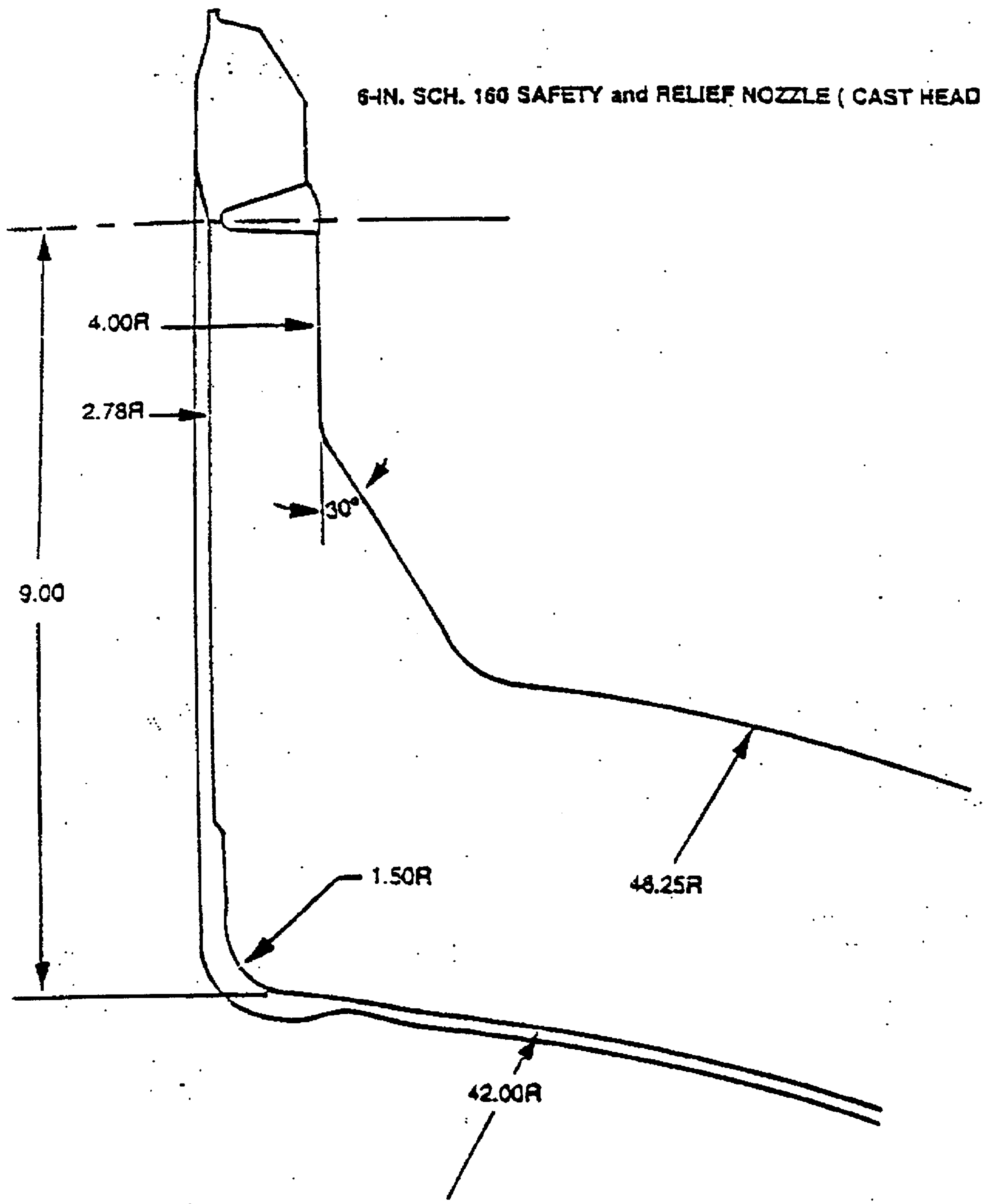


Figure 1-3 Geometry of a Typical Pressurizer Safety and Relief Nozzle

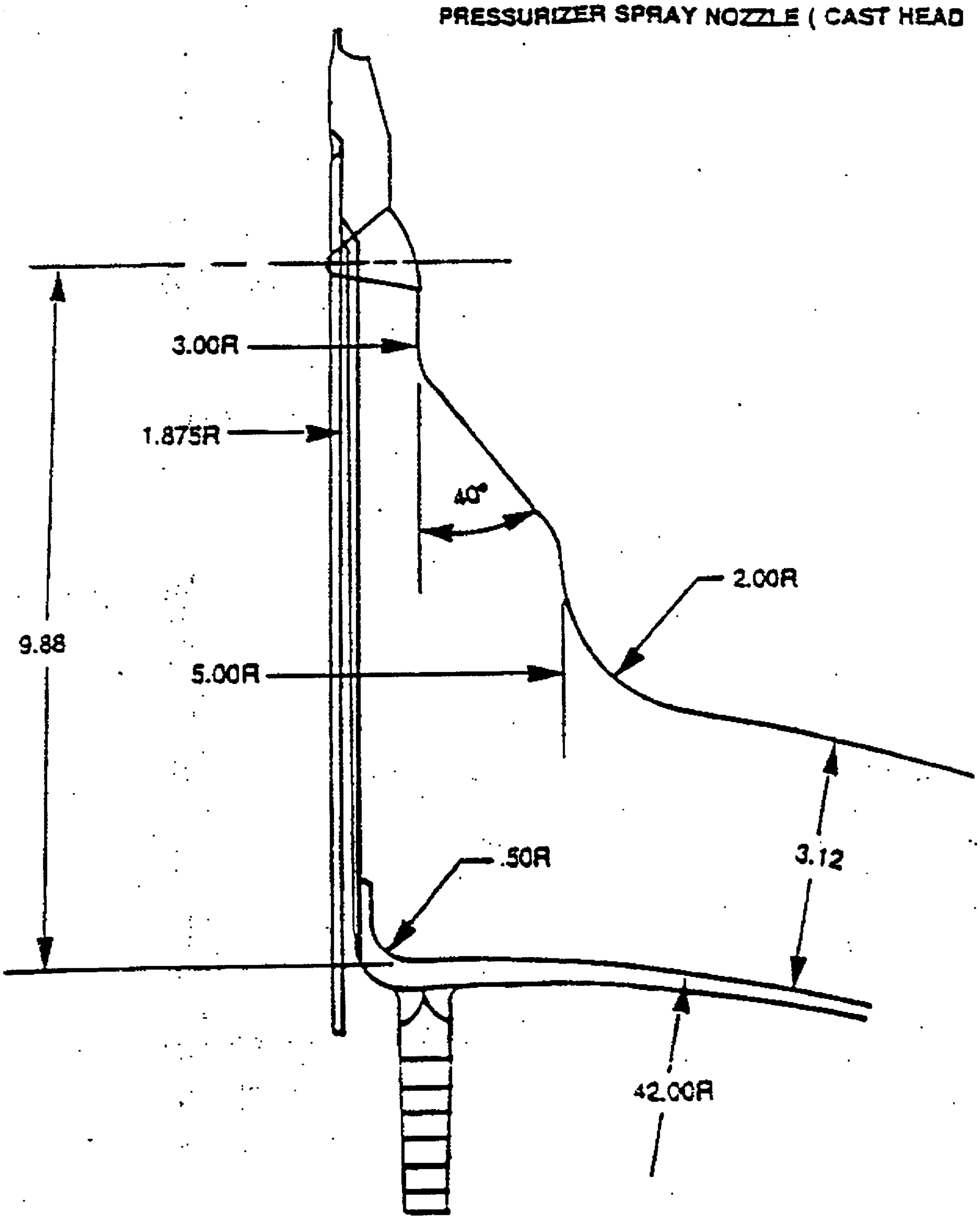


Figure 1-4 Geometry of a Typical Pressurizer Spray Nozzle - Cast Head Design

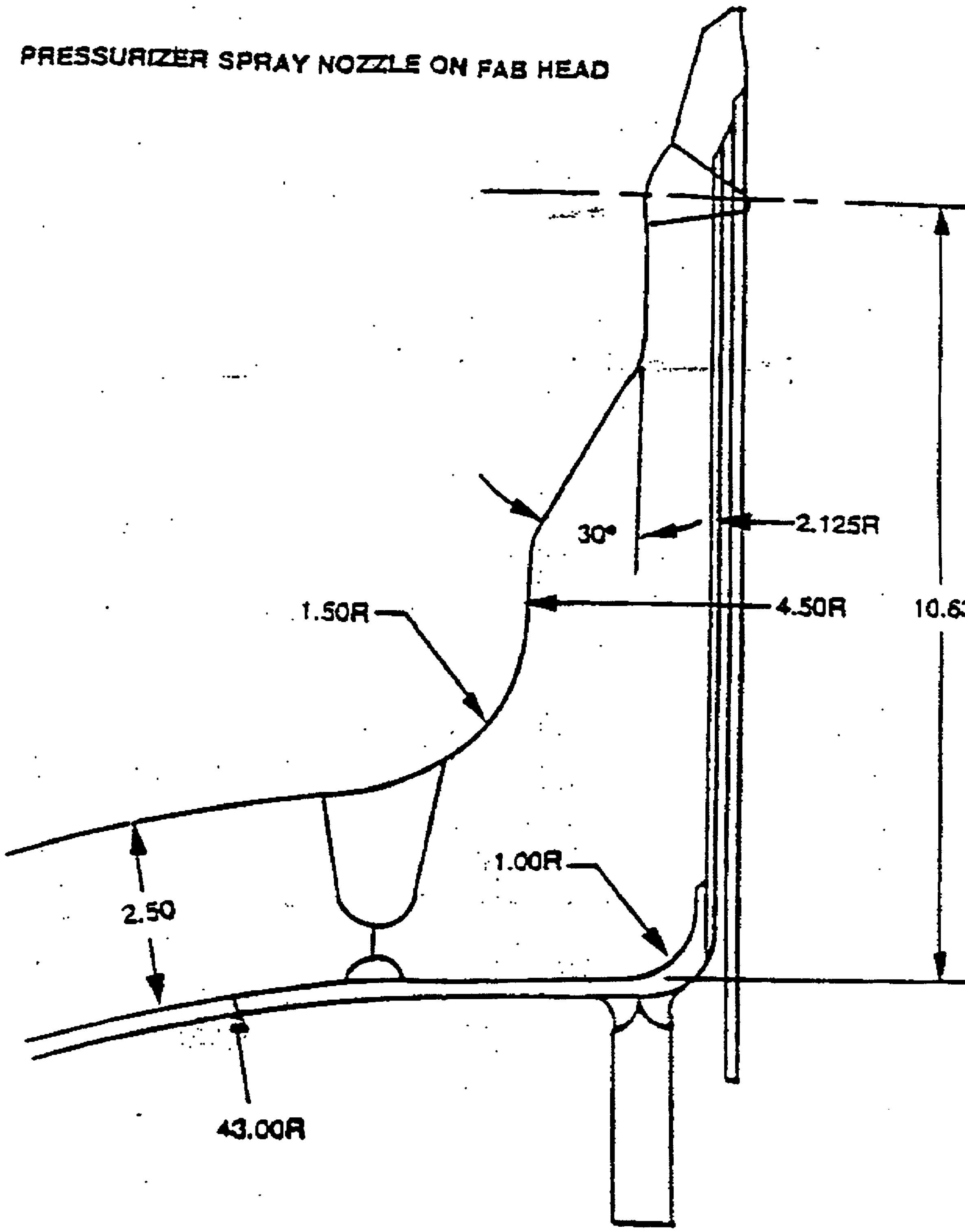


Figure 1-5 Geometry of a Typical Pressurizer Spray Nozzle - Fabricated Head.

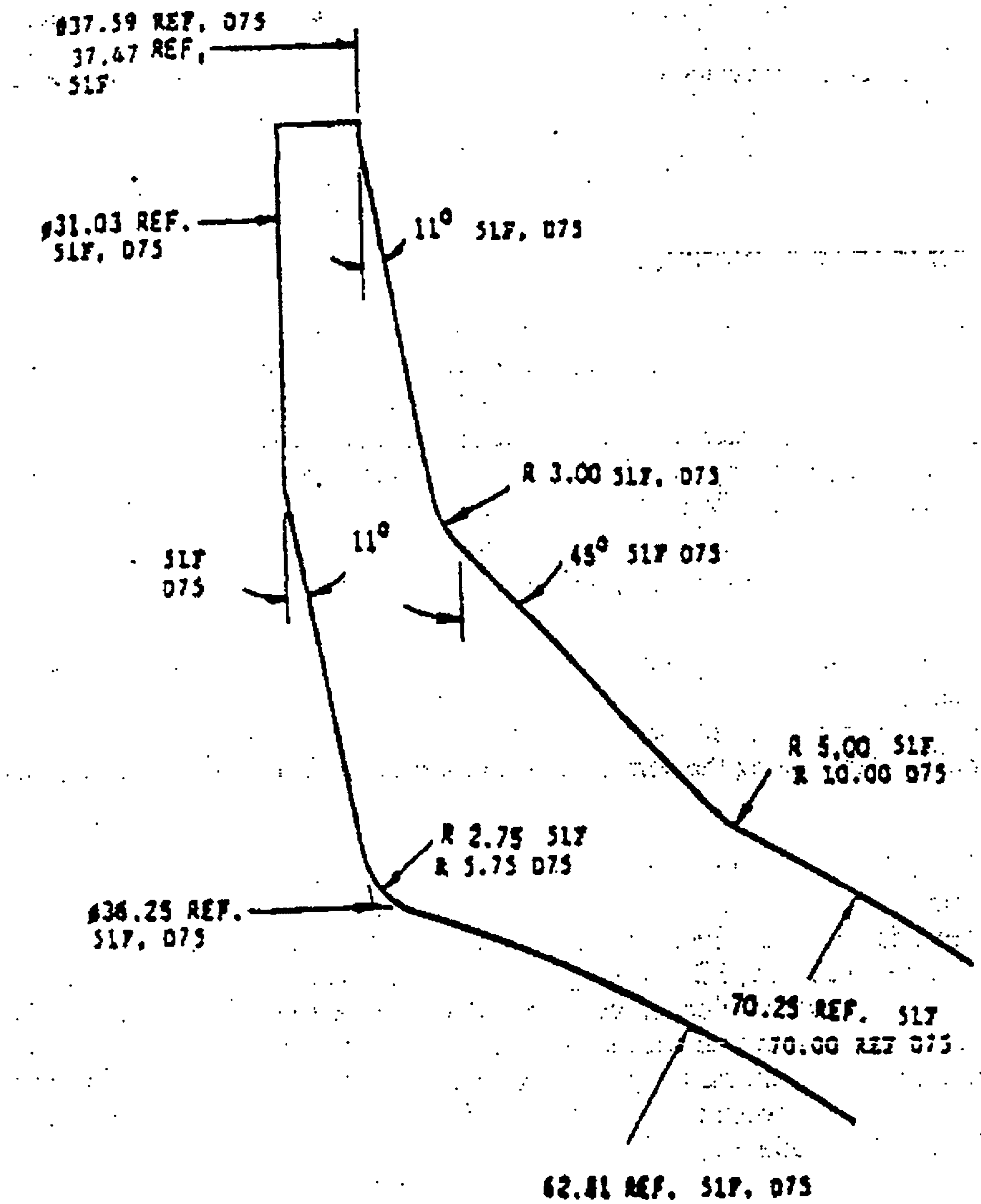


Figure 1-6 Geometry of a Typical Steam Generator Primary Nozzle.

2. Inspection History

The nozzle inner radius, as well as all the nozzle inner surfaces, are subjected to a surface examination both before and after the weld depositing of the stainless steel cladding. The inspection before cladding also includes a 100 percent volumetric exam, either UT or radiography, and the inspection after cladding is performed after the shop hydrotest including a radiographic exam for acceptance to ASME Section III requirements. This is generally followed by the baseline UT exam for Section XI.

2.1. Examination Volumes

Steam generator and pressurizer primary nozzle inner radius examination volumes are defined by the radius of curvature and the base metal thickness of the adjoining shell or dome plate. This requirement results in inspection areas that encompass the inner radius and the inside surface of the nozzle barrel. The inspection depth is 0.5" into the nozzle base metal excluding cladding. The flaw of interest is axial-radial in orientation, as depicted in Figure IWB-2500-7(b) (Figure 2-1).

2.2. Examination Approaches

Typically, access restrictions and radiological concerns preclude contact examination of the inner radii volume from the component interior. As a result, the standard approach is to perform contact ultrasonic examinations from the nozzle outside diameter surface radius blend, along the nozzle barrel and sometimes from the attached dome or shell plate.

The objective of all 3 scanning patterns is to provide complimentary coverage and completely interrogate the specified volume. The complexity of the examination effort depends on the geometric relationship between the outside surface and the inner radius volume. Recently, 3D modeling has been used to calculate ideal examination angles and predict the extent of coverage. Figure 2-2 shows a pressurizer safety or relief nozzle section view with beam coverages and recommended scanning patterns. These two nozzles have identical geometry. Figure 2-3 shows the pressurizer surge nozzle. Here the relationship between the O.D. (Outside Diameter) blend area and the inner radius volume is more favorable, requiring less exam complexity.

It is standard practice for utilities to approach primary nozzle inner radius examinations with specialized techniques designed to compliment the geometric configuration of the scanning surface. Examination procedures commonly specify contoured transducer wedges, special calibration blocks, and examination angles designed to intercept the inner radius corner at 45°.

2.3. Access and Exposure

For pressurizer safety, relief and spray nozzles, exams are usually performed from semi-permanent platforms at the elevation of the pressurizer upper head. Dose rates vary by plant but can be estimated at 100 MR/hr.

The pressurizer surge nozzle is not easily accessible so the exam surfaces are obstructed by lower head heater penetrations and the radiation fields are generally 200-300 HR/hr. Roughly half of all utilities surveyed have sought and received relief from volumetric examinations for those reasons.

The steam generator primary nozzle inner radius exams are not optimally accessed, and radiation fields can range from 100-300 MR/hr to 1R/hr. In addition to concerns regarding dose rate, this examination has been judged by many plant owners as complicated by material test problems as the integrally cast A216 channel head material (present in most channel heads) can complicate meaningful interpretation of ultrasonic data.

2.4. Inspection Results

A total of 25 utilities were surveyed in an effort to gain perspective on the state of primary nozzle inner radius examinations. From steam generator primary nozzle inner radius examinations, the survey population included 230 nozzles. From that population, 144 volumetric (U.T.) examinations have been performed or are planned. The remaining 88 nozzles in the population are visually inspected in the inner radius area. No *service induced* flaws have been detected in all examinations performed, as shown in Table 2-1.

The pressurizer surge nozzle has not been extensively examined by UT due to the access restrictions to the scanning surface on the outside diameter. Forty-eight percent of responding utilities have been granted relief from volumetric examinations, and the remaining utilities continue to attempt ultrasonic examinations of the inner radius from the O.D. surface. It is a safe assumption that nearly all examinations performed have had documented limitations. No service-induced flaws have been reported.

Pressurizer spray, safety and relief nozzles are generally accessible and the survey indicates a high percentage of volumetric examinations are being performed (159 nozzles, 146 U.T. examinations, 13 visual examinations). No service-induced flaws have been reported.

TABLE 2-1 INSPECTION RESULTS

Nozzles	Total Inspections	Indications
Steam Generator Inlet & Outlet	291	0
Pressurizer Spray	63	0
Safety Injection	4	0
Relief Nozzle, Safety Nozzle	122	0
Pressurizer Surge	26	0

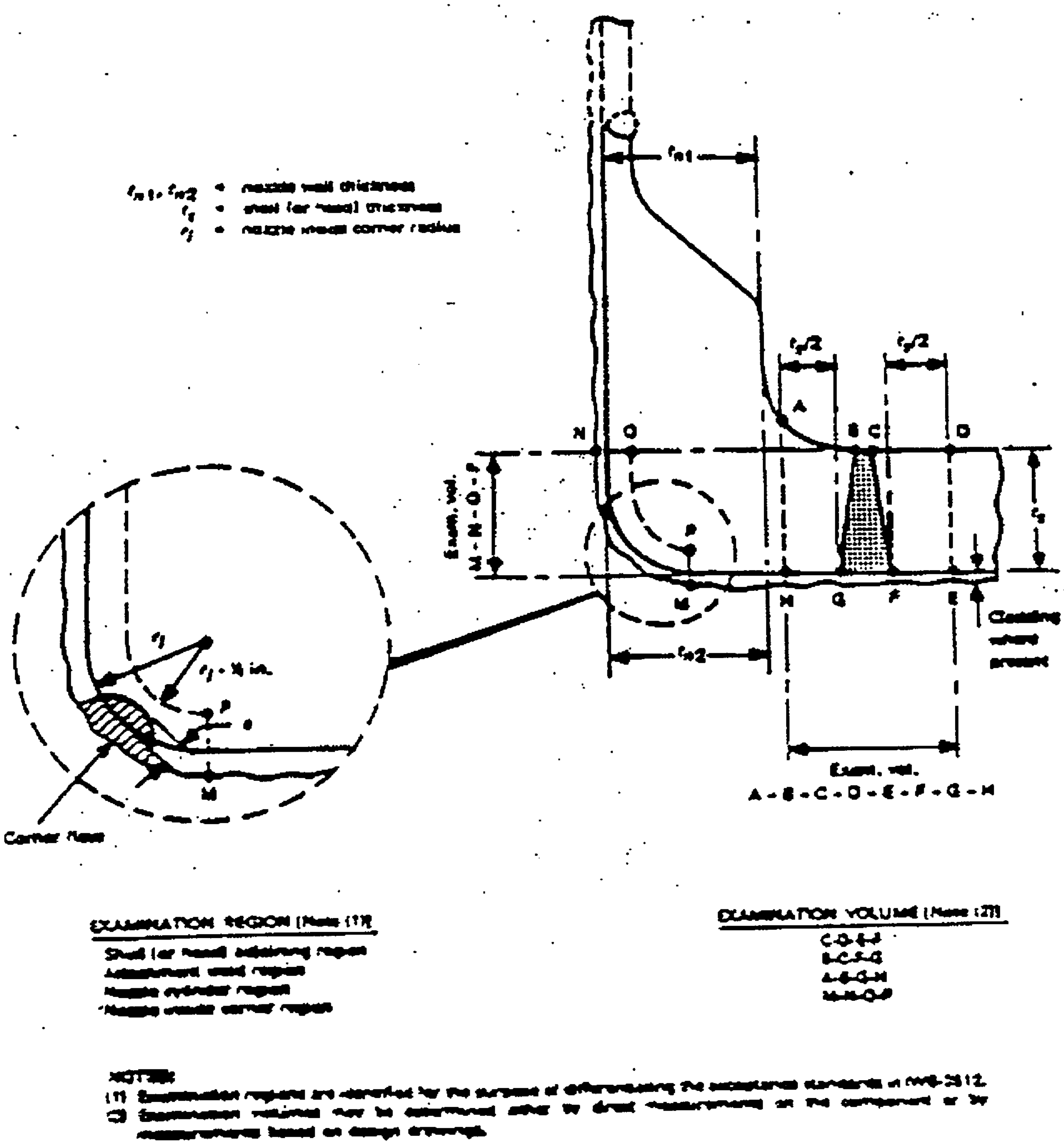


Figure 2-1 Examination Volume for a Nozzle Inner Radius,
from Section XI, Figure IWB 2500-7(b)

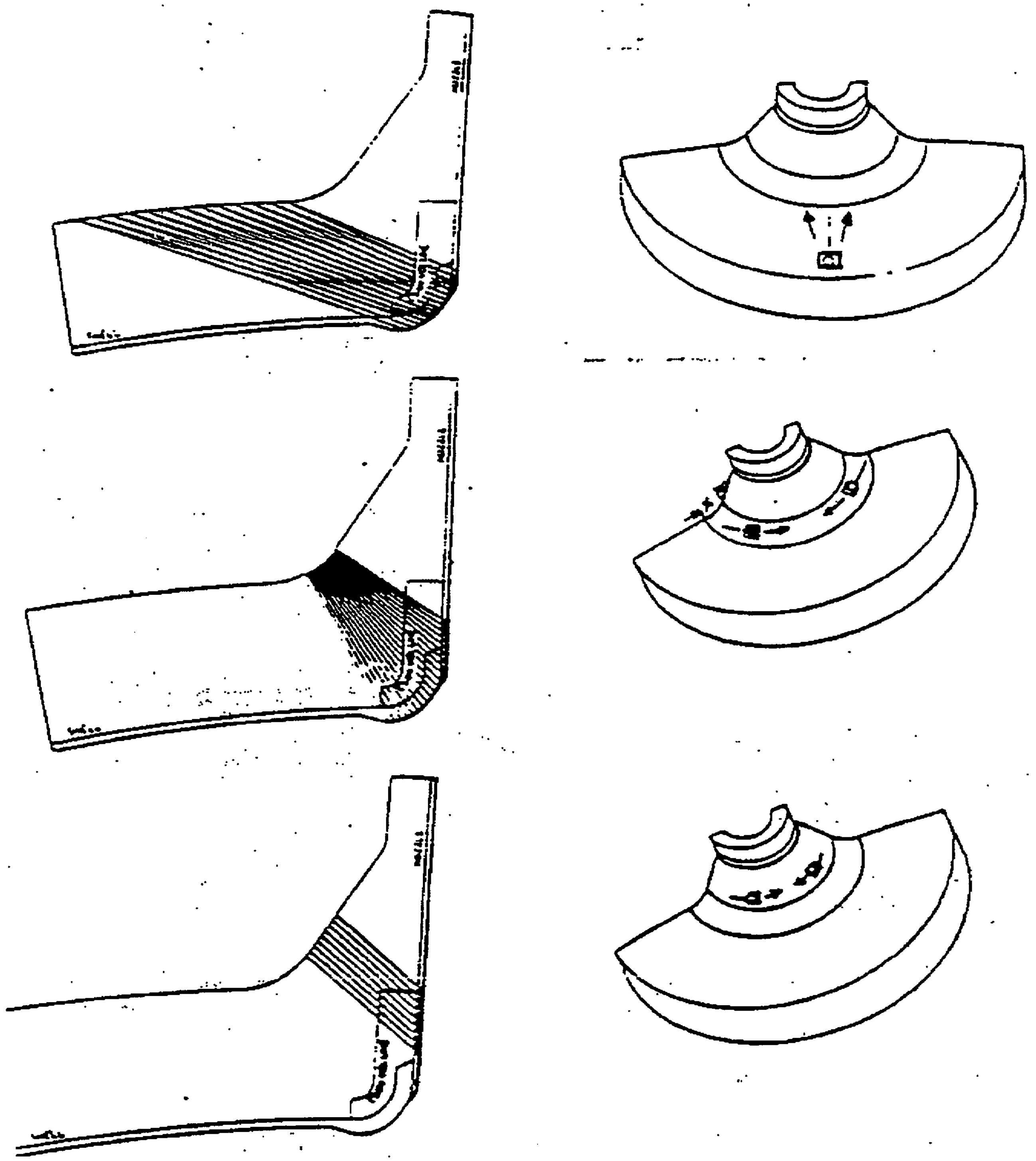


Figure 2-2 Scan Paths and Examination Volumes for a Typical Pressurizer Safety or Relief Nozzle

3. Fracture Assessment – Deterministic Approach

Five different nozzle geometries were evaluated by fracture mechanics assessment to determine the stress intensity factors for various postulated crack sizes. The nozzles evaluated are the pressurizer fabricated surge nozzle, the pressurizer spray fabricated and cast nozzles, the pressurizer safety and relief cast nozzle.

Finite element analyses were performed to obtain stresses at the nozzle inner radius regions due to all the design thermal transients. The maximum stress profiles obtained were used to calculate the stress intensity factors (K_I). The magnitude of stress intensity factor depends on the distribution of the applied stress and the geometry of the crack and the structural component. The stress intensity factor is the driving force for crack extension caused by the applied stresses. In the Linear Elastic Fracture Mechanics (LEFM) theory, the stress intensity factor is a single parameter that characterizes the stress and strain distributions in the immediate vicinity of the crack tip. The material resistance to crack propagation is called fracture toughness, K_{IC}. Within the regime of LEFM, crack initiation and fatigue crack growth can be predicted in terms of the stress intensity factor.

Two methodologies were used to determine the stress intensity factors. One method was developed [1] for a semi-circular crack in a nozzle corner. The stress intensity factor is calculated by

$$K_I = \sqrt{\pi a}\left[0.706A_0 + 0.537\left(\frac{2a}{\pi}\right)A_1 + 0.448\left(\frac{a^2}{2\pi}\right)A_2 + 0.393\left(\frac{4a^3}{3\pi}\right)A_3\right] \qquad (3\text{-}1)$$

where a = crack depth, A_0, A_1, A_2, and A_3 are stress coefficients, representing the far field stress distribution normal to the crack plane, as defined below.

$$\sigma = A_0 + A_1x + A_2x^2 + A_3x^3 \qquad (3\text{-}2)$$

where x is the distance from the surface (into the wall thickness) where a crack is assumed to begin propagating. Eq. (3-1) has been used by EPRI [1] for nozzle corner cracks subjected to combined thermal and pressure stresses.

As shown in [1], the K_I solution for a nozzle corner crack is very comparable to the K_I solution for a semi-circular crack in half-space and a quarter circular crack in quarter-space solids. All these 3 classes of cracks assume a semi-circular flaw.

The other method was developed by Raju and Newman [2]. This method covers a wide variety of flaw shapes. The cracks can be assumed either on the inside or the outside surface of a cylinder with various ratio of thickness to inside radius. The Raju-Newman method is more versatile and could lead to more conservative results by assuming greater aspect ratios. The results shown in the following section provide a comparison between the two methods, with aspect ratio 6 used in the Raju-Newman calculations. It should be noted that cracks originating from the nozzle corner are more likely restricted to smaller aspect ratios, typically, R=2. With R=2, which is what is embedded in the method of [1], it can be shown that the Raju-Newman method and the ref. [1] method are very comparable.

3.1. Fracture Toughness and failure criteria

In ASME Section XI, Appendix A, there are two fracture toughness equations available for fracture evaluation [1]

$$K_{Ia} = 26.8 + 1.233 \exp\left[0.0145\left(T - RT_{NDT} + 160\ ^{\circ}F\right)\right] \qquad (3\text{-}3a)$$

$$K_{IC} = 33.2 + 2.806 \exp\left[0.02\left(T - RT_{NDT} + 100\ ^{\circ}F\right)\right] \qquad (3\text{-}3b)$$

where T is the temperature of the structural components and RT_{NDT} is the reference temperature of nil ductility of the material. K_{Ia} is the dynamic fracture toughness used for crack arrest criterion and K_{IC} is the static fracture toughness used for crack initiation criterion. The unit for K, K_{Ia}, and K_{IC} used in the entire report are $ksi\sqrt{in}$.

Equations 3-3a and 3-3b are bounded by the upper shelf value of 200 $ksi\sqrt{in}$. Different fracture criteria are used for different plant conditions, and are listed below:

$K_I \le \dfrac{K_{Ia}}{\sqrt{10}}$ (Normal, Upset and Test)

$K_I \le \dfrac{K_{IC}}{\sqrt{2}}$ (Faulted or Emergency)

Therefore, for normal and upset conditions, the allowable flaw size per Section XI can be determined by using a reduced toughness of $200/\sqrt{10} = 63.2\ ksi\sqrt{in}$ since the nozzles are operating at temperatures above 300 °F where the upper shelf toughness prevails.

3.2. Fatigue Cack Gowth

Fatigue crack growth may be estimated by

$$\frac{\Delta a}{\Delta N} = C(\Delta K)^n \qquad (3\text{-}4)$$

where $\Delta a/\Delta N$ = fatigue crack growth rate, ΔK = stress intensity factor range, and C and n are material properties. Eq. 3-4 can be easily evaluated to estimate fatigue crack growth, Δa, by integration within a time period in which the ΔK remains relatively constant. Integration continues with an updated crack depth and new ΔK.

The ASME fatigue crack growth data which will be used are shown in Fig. 3-6. Deterministic fatigue crack growth evaluations were performed and show very small amounts of growth in the entire operating life. Crack growth will also be covered in the risk evaluations described in the following section.

3.3. Results of Deterministic Facture Evaluations

Fracture evaluations were performed for the pressurizer nozzles and steam generator primary nozzles. Both the Besuner [1] and the Raju-Newman methods were used to calculate the stress intensity factor. The maximum allowable crack depths were determined using the fracture criteria described above. The K_I versus the normalized crack depth, a/t, results are shown in Figs. 3-1 to 3-5. The allowable a/t values are summarized in Table 3-1.

Table 3-1 . Analysis Results

Nozzle Name	Manuf. Method	Figure No.	Thickness	Critical Flaw Depth (a/t) (Besuner)	(Raju-Newman)
Pressurizer Surge	Fabricated	3-1	3.577"	> 0.9	> 0.9
Pressurizer Safety & Relief	Cast	3-2	4.764"	> 0.9	> 0.9
Pressurizer Spray	Cast	3-3	4.459"	> 0.9	> 0.9
Pressurizer Spray	Fabricated	3-4	3.289"	> 0.9	> 0.9
Steam Generator Primary Nozzle	Fabricated	3-5	10.237"	> 0.9	> 0.9

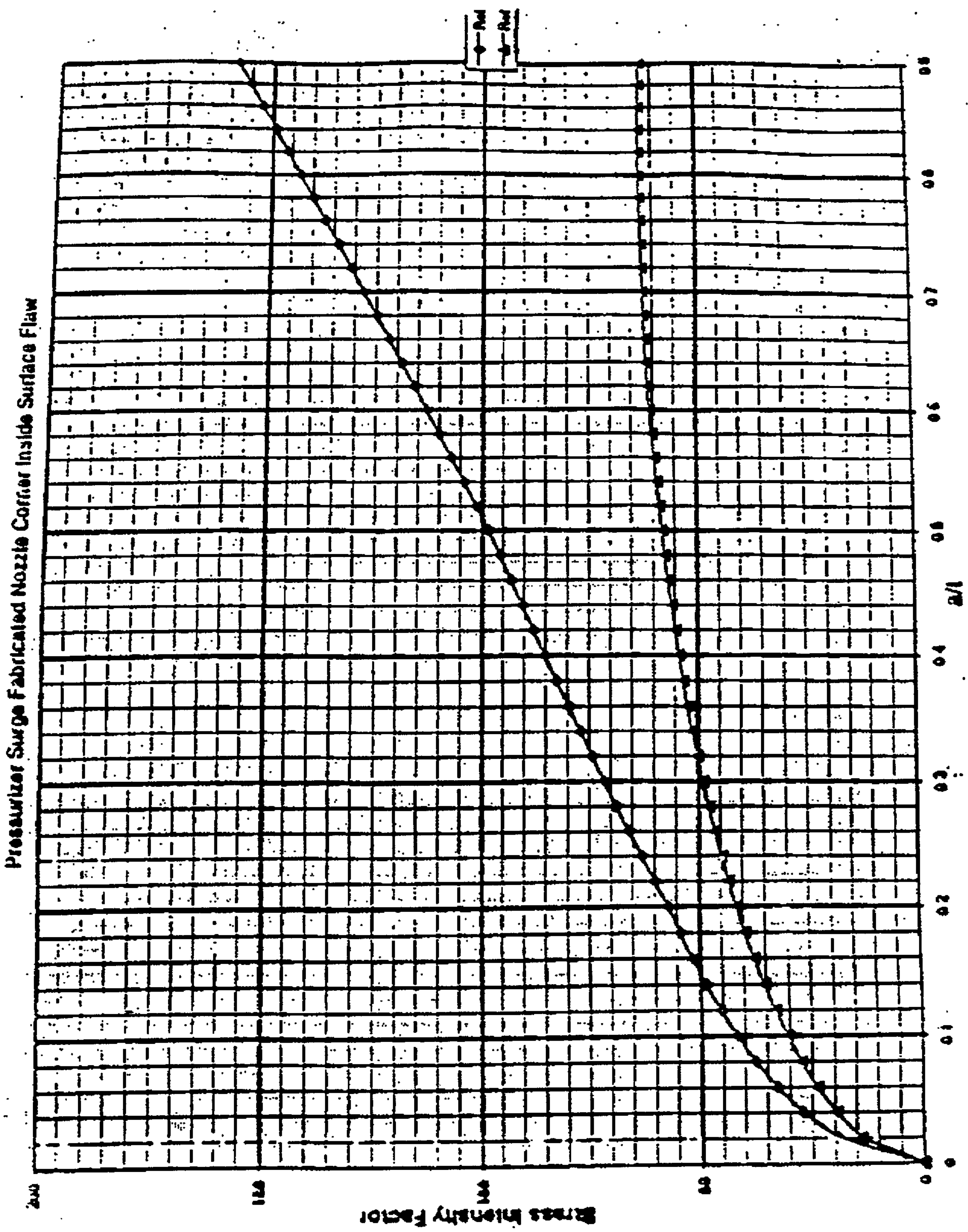


Figure 3-1 Fracture Evaluation Results - Pressurizer Surge Fabricated Nozzle

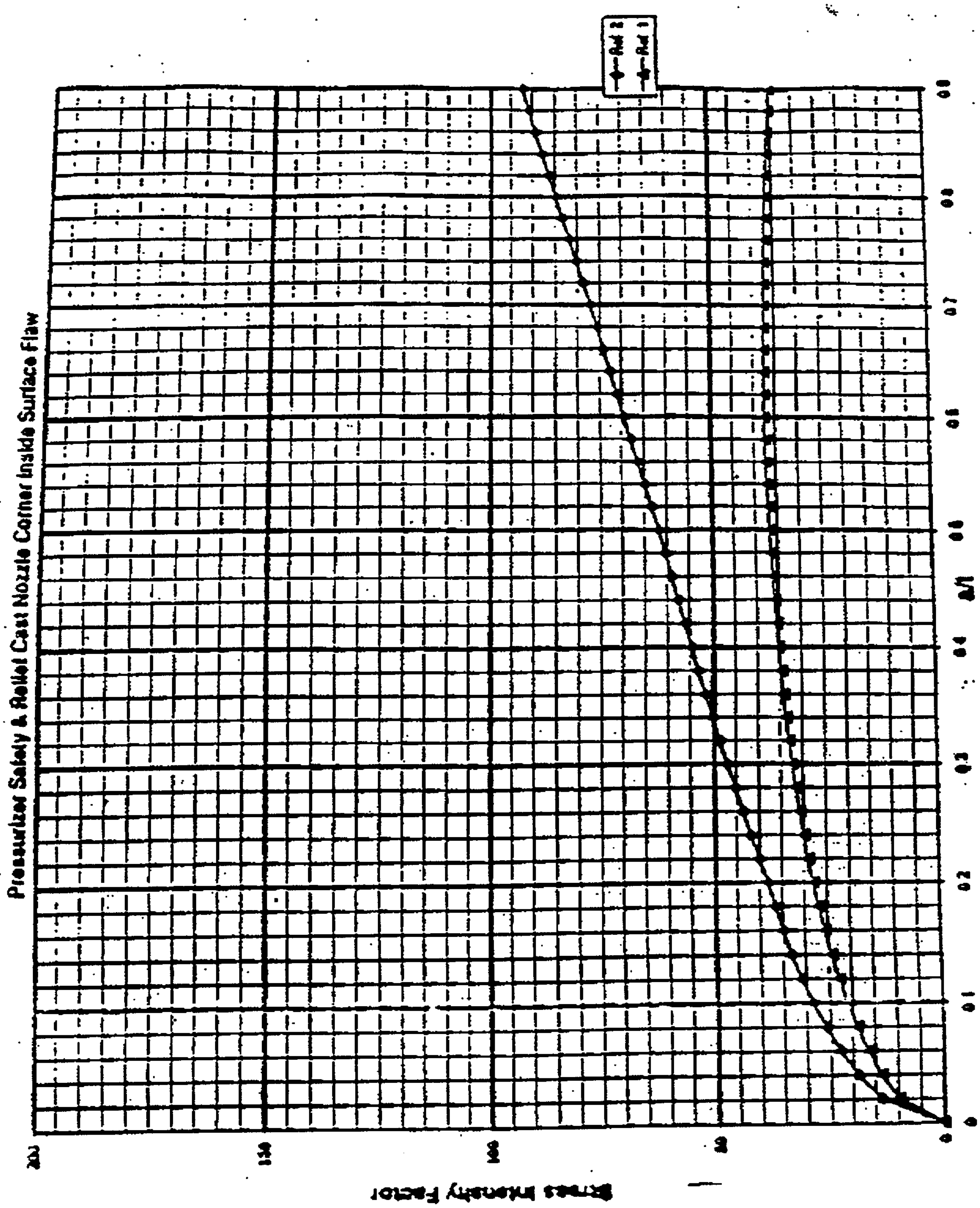


Figure 3-2 Fracture Evaluation Results - Pressurizer Safety and Relief Cast Nozzle

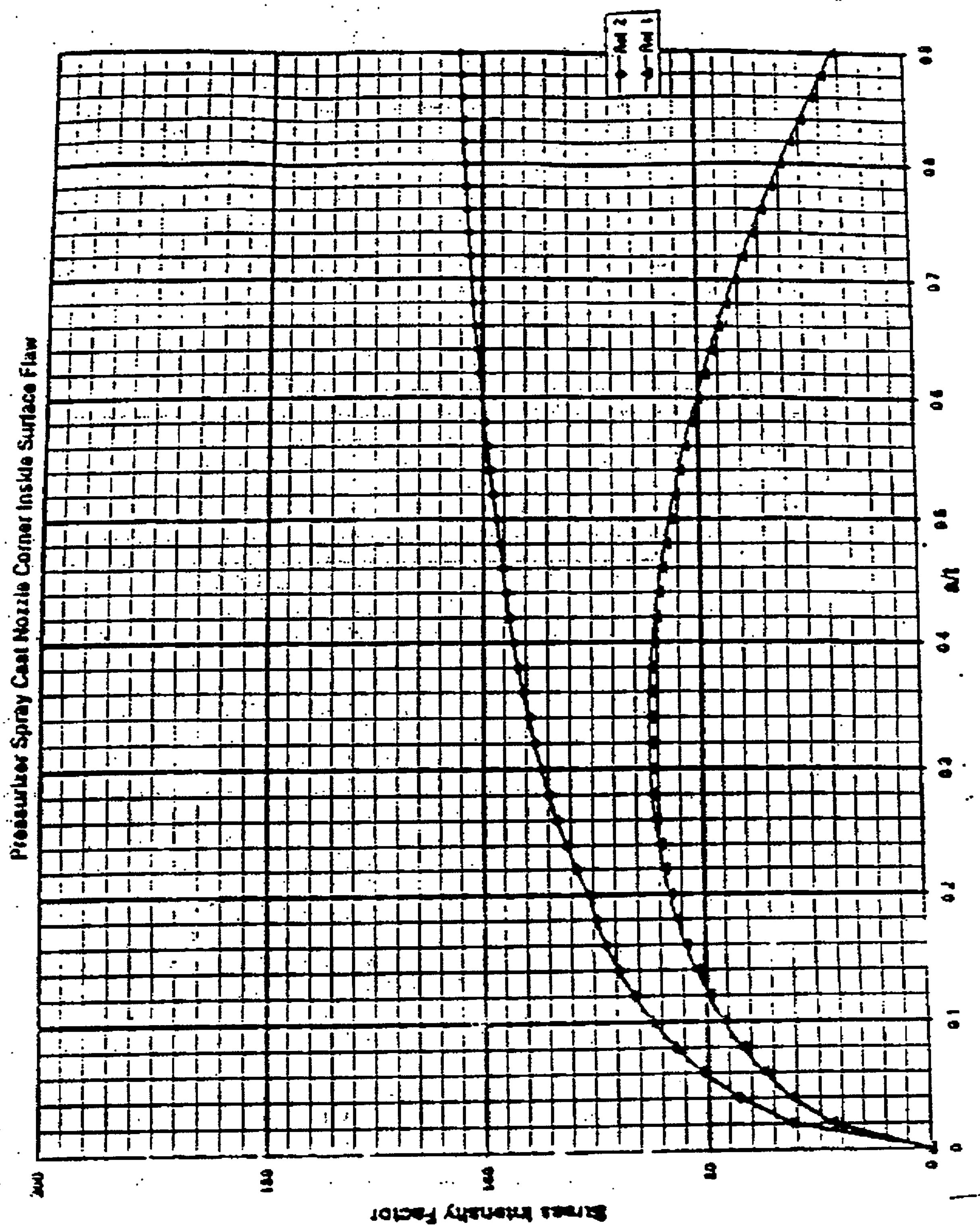


Figure 3-3 Fracture Evaluation Results - Pressurizer Spray Cast Nozzle

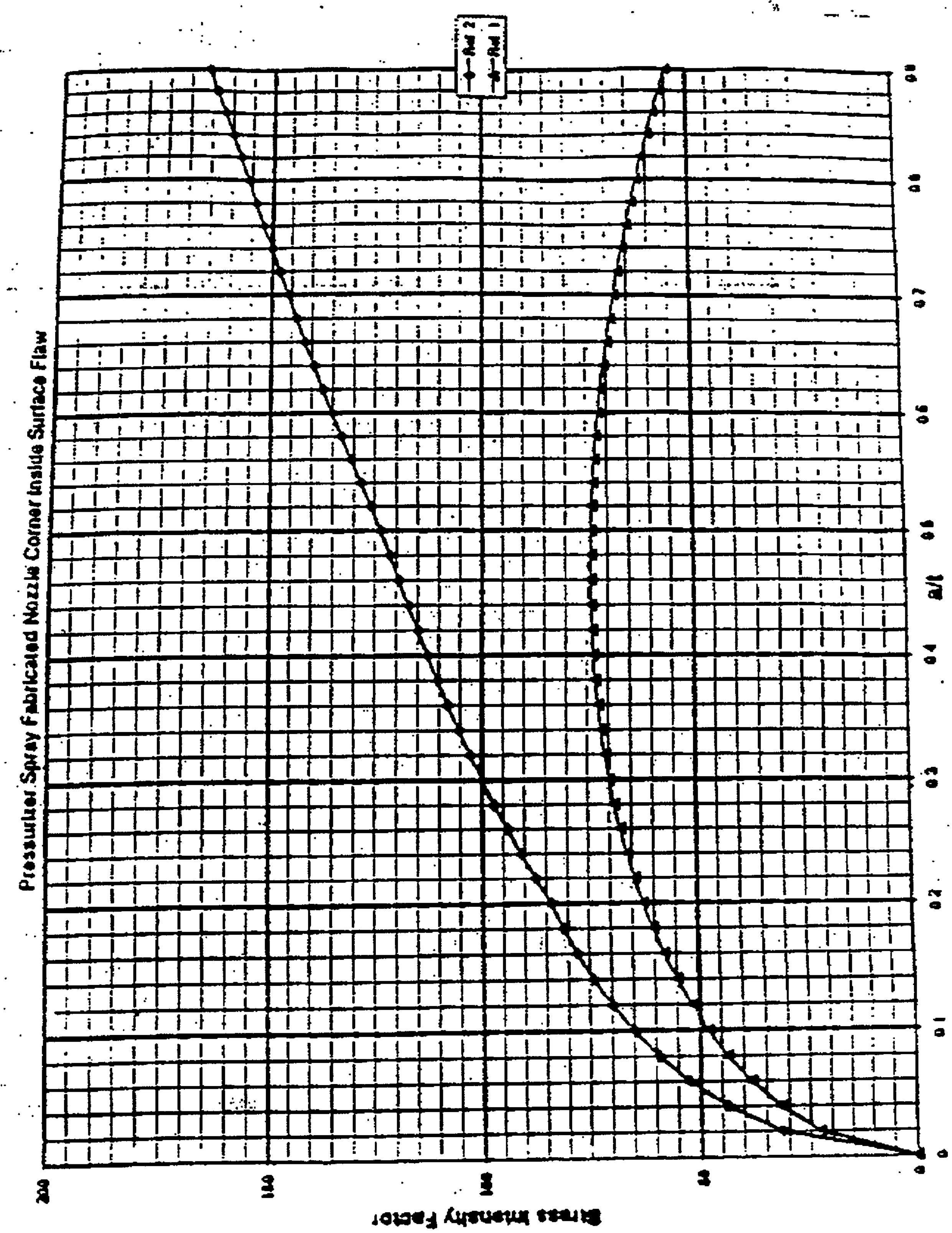


Figure 3-4 Fracture Evaluation Results - Pressurizer Spray Fabricated Nozzle.

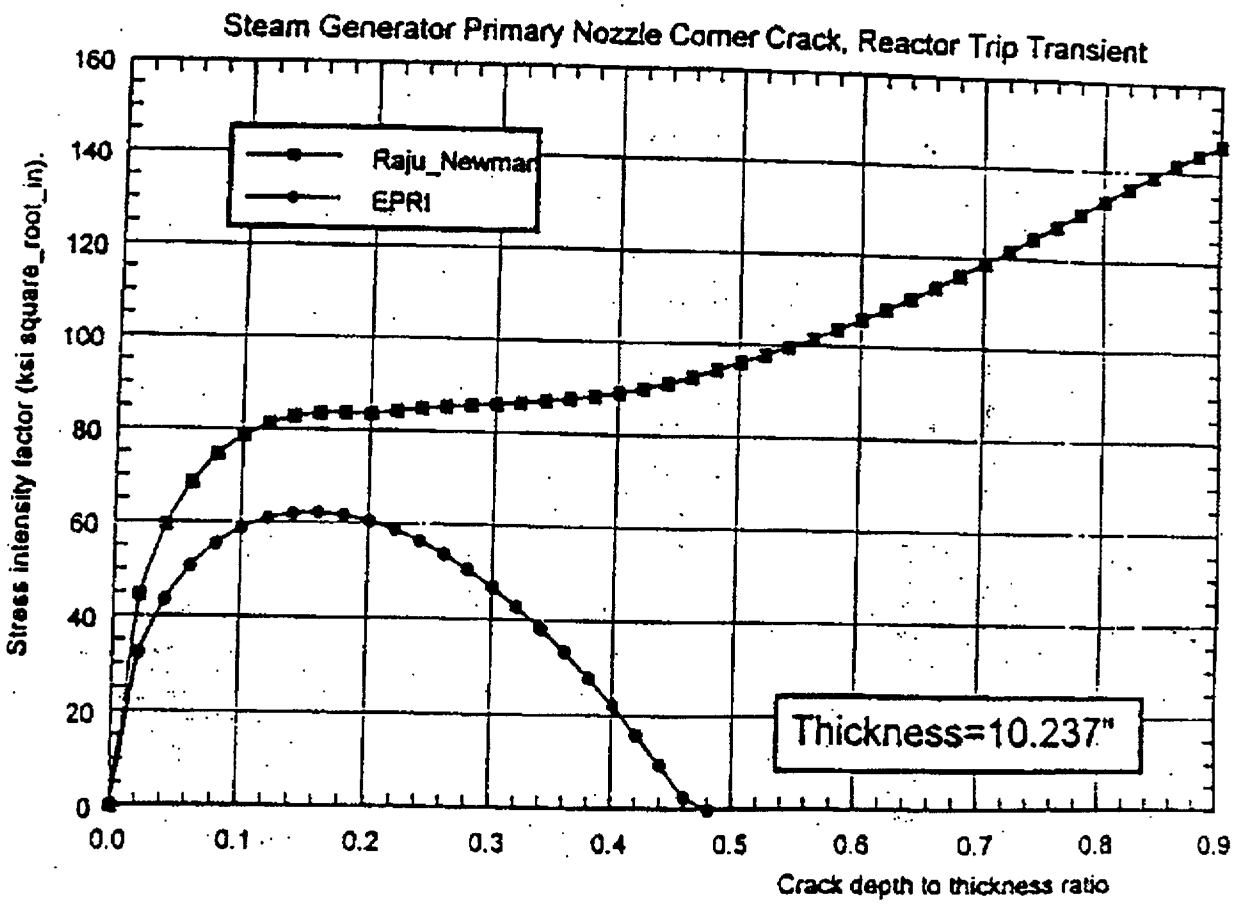


Figure 3-5 Fracture Evaluation Results - Steam Generator Primary Nozzle

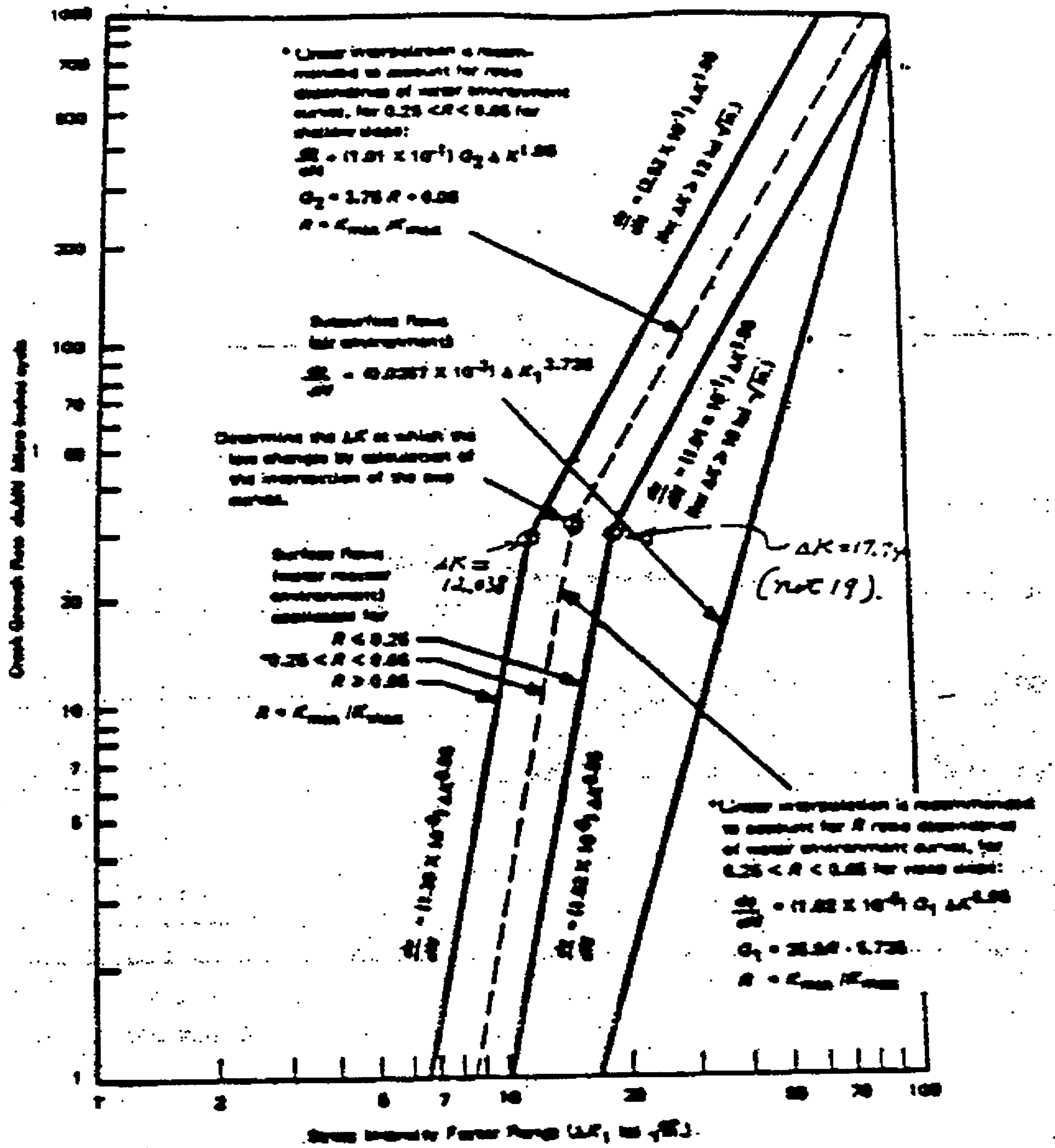


Figure 3-6 The ASME Fatigue Crack Growth Data for Low Alloy Steels in Both the PWR Water and Air Environments

4. Risk Assessment – Probabilistic Approach

In this section we evaluate the effects of in-service examinations on the risk of failure due to cracking in the nozzle inner radius. Since the applied stress intensity factor does not exceed the fracture toughness, it could be argued that leakage would occur from a through wall flaw before any integrity problems would occur, at any of these nozzles.

The key question to address in the risk assessment is whether in-service inspection can change the risk of failure by identifying in-service flaws.

There are no mechanisms of damage other than fatigue for the nozzle corners. Therefore, the only scenarios of concern are for a flaw which was not found in the pre-service examination to grow during service, or for a flaw to initiate during service and propagate.

The nozzles have all been examined by both UT and MT (magnetic particle testing) prior to the cladding being applied, per the requirements of the material specification. After cladding, the nozzles were required to be liquid penetrant tested to ensure the integrity of the cladding. With these examinations, the probability of non-detection for the pre-service cracks is very low.

4.1. A Brief Description of the Risk Assessment Methodology

The risk assessment employed the Monte Carlo method to determine probability of failure accounting for the statistical aspects of the relevant physical quantities. If the number of trials, N, is sufficiently large, the probability of failure, P_f, approaches the ratio of the number of samples that are failed, N_f, to N, namely,

$$P_f = \frac{N_f}{N} \qquad (4\text{-}1)$$

Note that Eq. 4-1 may be weighed using importance sampling and probability of non-detection, etc. The outcome of P_f depends on the applied loads, and material properties which are treated as random variables with specific statistical distributions. Inspection and repair can also affect the outcome. Multiple failure mechanisms can be included in the evaluation. For the present application, however, as mentioned above, fatigue crack growth is the only cracking mechanism considered possible.

Within each trial, fatigue crack growth is calculated and accumulated for all years over the plant design life. Failure criteria are checked at the end of each year and the in-service inspections are performed according to the schedule. This process repeats for all trials, each with a new set of random variables which simulates various conditions under which fatigue crack growth might occur. Through the trials the failed cases are identified and accumulated and the non-detection probability modified after each inspection. Finally, Equation 4-1 is evaluated, after weighing with the importance sampling and the probability of non-detection factors relating to in-service inspections, to determine the failure probability.

Analyses were performed for the Surge, Spray, Safety Relief, and Steam Generator Primary Nozzles. Note that only the fabricated nozzles were evaluated because they are more limiting than the cast nozzles

4.2. Flaw Depth Distributions

Studies have shown that the distribution of flaws in reactor pressure vessels follows the Marshall distribution [3]. The initial flaw depth distribution is the Marshall distribution without the effect of in-service inspection [3], with the following cumulative probability function

$$F(x) = P(a \leq x) = 1 - \exp(-4.06x) \qquad (4\text{-}2)$$

where x is the crack depth. This is a special case of the Weibull distribution whose cumulative probability function is

$$F(x) = P_x = 1 - \exp\left[-\left(\frac{x}{\beta}\right)^{\alpha}\right] \qquad (4\text{-}3)$$

with $\alpha=1$ and $\beta=1/4.06$.

4.3. Analysis for the Nozzles

Analyses for the Safety Relief Nozzle, Spray Nozzle, Surge Nozzle, and Steam Generator Primary Nozzles were performed. A normal distribution was assumed for the C-constants, with the standard deviation = 10% to 20% of the mean. The n-exponents are assumed to be fixed constants. The failure probability results are shown in Table 4-1. The failure probabilities are very low, and the effect of inspection is very small.

Table 4-1 Results of Evaluations

Nozzle type	Probability* of failure (Without Inspection)	Probability* of failure (With Inspection)
Relief Nozzle	3.28×10^{-8}	3.07×10^{-8}
Spray Nozzle	6.96×10^{-6}	9.12×10^{-8}
Surge Nozzle	2.02×10^{-9}	5.10×10^{-10}
Steam Generator Primary Nozzles	6.55×10^{-6}	1.58×10^{-6}

* Number of trials = 25000, with importance sampling.

5. Discussion

The analysis results shown in this report are based on the conservative assumptions and data. Extremely small failure probabilities were obtained based on these conservative calculations. The initial flaw depths used in the analyses were also very conservative. Per the studies of Fred Simonen, Battelle, Pacific Northwest National Laboratory (PNNL), most of the flaws found in destructive examination of reactor pressure vessels are smaller than 0.08" [3]. The Marshall distribution used in the present analysis used flaws with considerable initial depths for evaluation. About 7% of the trials had initial flaw depth greater than 0.65" and 1.2% of the trials had initial flaw depth greater than 1.0".

The benefit of in-service inspection is negligible. Table 4-1 shows that there is about 2 orders of magnitude difference between the two evaluations: with inspection, and without inspection. Since the probabilities are so small, the gain is meaningless.

6. Conclusions

The results shown in this report have demonstrated that it is highly unlikely that the nozzles considered in this report would fail under any anticipated service conditions. In-service inspections can hardly benefit plant safety for something that is very unlikely to happen. The inspection is very difficult to perform because of access, and the high radiation environment in many cases. Inspections which have been done have not led to discovery of any indications at all. It is recommended that in-service inspections on all PWR pressurizer and steam generator nozzle inner radius regions be eliminated for economic and health reasons, without any risk to structural integrity.