UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SECRETARIAT RECORD COPY

APR 26 1979

MEMORANDUM FOR: Commissioner R. T. Kennedy

THRU: Lee V. Gossick, Executive Director for Operations

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: SEISMIC EVALUATIONS OF FIVE NUCLEAR POWER PLANTS

This is in response to your memorandum of March 14, 1979.

The seismic analysis methods for the five affected plants were reviewed in some detail, especially at the OL stage of review, and found to be acceptable. However, the staff in its review did not explore the spatial (intramodal) method of combination used in the dynamic analysis of system piping. The review was sufficient to disclose that acceptable methods were used in combining modal responses, but we can find no indication in the agency records that the intramodal method of combination was described or questioned on any of the five plants. Records for other plants of Stone and Webster design that we have reviewed in recent years do contain descriptions of acceptable methods for these spatial response combinations. In addition, the enclosed describes the present state of verification of stress analysis methods.

A brief description of how our seismic design methods have evolved follows. In the early years of nuclear regulation, prior to 1967, there were no formal regulations or guidance on seismic design methods. The state of the art of seismic design during this time was perhaps best described in a document entitled "Nuclear Reactors and Earthquakes" (TID-7024) issued in August 1963, by the U.S. Atomic Energy Commission. The report reflected the practices employed in the design of government-owned reactors at that time. Applicants for AEC licenses were made aware of the existence of such documents and instructed to employ them in design of their nuclear power plants. The methods used for seismic design in the period prior to 1967 were the so-called equivalent static methods.

In the equivalent static load method of analysis a single static force is applied at the center of gravity of the structure or component. In using the method, the designers usually took the peak of the calculated dynamic response of the structure, multiplied

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it by some factor between 1.5 and 2, and then calculated an equivalent static force. This single force was intended to represent the forces due to the inertia of the structure and the amplification of those forces due to the dynamic nature of the loading. Although this approach is suitable for systems of simple geometry, it was found to possibly underestimate the seismic response of complex systems in some cases and to overestimate in others.

Starting about 1967, various experts in the field of seismic design, most notably Dr. Nathan Newmark at the University of Illinois, published papers demonstrating that advanced dynamic analysis techniques that were a technological spinoff from the aerospace industry could be applied to the seismic design of structures. The application of these advances in the state of the art to nuclear plant design was encouraged and supported by the AEC regulatory staff because they permitted better characterization of the actual response of nuclear power plant structures and systems to an earthquake. It is also important to note that the use of these more advanced dynamic analysis techniques in design of complex structures like nuclear power plants was feasible by the late 1960s because of the increasing availability of computers with sufficient capacity and calculating speed.

When the staff began to require dynamic analysis in the design of structures and components for seismic loading in about 1967, the methods and practices employed by industry were based on the available technical literature and on what had evolved as accepted engineering practice in the field of dynamic analysis as it was appplied outside the nuclear industry. Inherent in the dynamic analysis techniques was the recognition that actual structures and systems would respond to an earthquake in several simultaneous modes of vibration. This meant that a mathematical method was necessary for combining the spatial (intramodal) components of the seismic response at a given point in a structure or system to determine the total response. However, the regulatory staff guidance on acceptable techniques of dynamic analysis for use in license applications was limited to basic criteria such as earthquake and accident loading combinations, allowable stress and deformation limits and damping values. These criteria were communicated principally through the question and answer process used in the staff review of an application. The NRC records disclose

that no criteria were issued at the detailed level of analysis involving the combination of spatial response components in piping or structures in these early years.

Beginning about 1967, consulting organizations were retained by the AEC regulatory staff to assist in the evaluation of seismic design criteria for most plants, including Maine Yankee, Surry, Fitzpatrick, and Beaver Valley. Expert and nationally recognized consultants were retained under contract with the AEC regulatory staff in lieu of hiring staff members with comparable expertise.

In the period 1970-1974 the staff was enlarged to include personnel with expertise in dynamic analysis, and a number of consultants were employed to assist in defining more specific requirements for seismic analysis. During this same period of time there was a great deal of activity in the engineering community in the development of techniques for dynamic analysis of nuclear power plants. A number of studies were undertaken by engineers in both academic and industrial circles to define the applicability and limitations of the analytical techniques that were coming into use, including the subjects of modal and spatial response combinations. From our regulatory point of view, this period culminated when the essence of these efforts was codified in NRC Regulatory Guide 1.92 "Components of Modes and Spatial Components in Seismic Response Analyses" first published in 1974 and revised in 1976. The guide is now in routine use in the licensing process and treats fully the method of response combinations of concern in the five affected plants.

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
"Present State of Verification of Stress Analysis Methods"

cc: Chairman Hendrie
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
A. Kenneke, OPE
L. Bickwit, OGC
S. Chilk, SECY
C. Kammerer, OCA
J. Fouchard, OPA

Present State of Verification of Stress Analysis Methods

Existing detailed requirements contained in pertinent Standard Review Plans and Regulatory Guides issued since the five plants were designed and approved have greatly reduced the chances that design errors of this type will take place. The Standard Review Plan sections and the Regulatory Guides which pertain to seismic analysis require a dynamic analysis, and provide for input time histories, ground response spectra, damping, modelling of structures, development of floor response spectra, and methods of combination of both spatial components and modal contributions. The Standard Review Plan also requires that applicants verify their dynamic analysis programs by comparison of results with those of other programs and with generally accepted solutions to benchmark problems. These current criteria are adequate and do not require change. Had they been in place at the time these five plants were reviewed, the error we are now concerned with would probably have been discovered.

To improve our confidence in computer results, the staff has for some time been in the process of establishing a standardized program for independently evaluating and verifying the quality of computer programs used for dynamic and static structural analysis of nuclear piping systems and components. This program consists mainly in the definition and solution of a set of standardized benchmark problems involving the analysis of a set of structures of progressively increasing complexity, representing typical piping system analyses

as found in currently proposed or operating plants. Increased assurance of proper code verification will be provided by requesting applicants to provide solutions generated with their computer programs to these standardized benchmark problems, and comparing these responses with the benchmark solutions. Agreement or deviation of results will provide an index of the adequacy and quality of an applicant's analysis methods. This program will also provide the NRC with the capability to perform independent calculations to verify applicants' dynamic analyses for particular designs.

The following paragraphs elaborate on the past and present staff efforts in the area of stress analysis code review and verification.

In 1973, the staff realized that there was a proliferation of computer programs for stress analysis, all of which would be required to be examined in the process of licensing reviews. Due to the substantial number of plants under review at that time, it was decided that a generic program to review these computer programs should be instituted that would have two goals:

1. To provide independent in-depth verification of the capabilities of the programs claimed by the applicants in the SARs; and
2. To provide the staff with a list of acceptable computer programs that would reduce the review effort in at least one area.

In February 1974, an outline for a validation program was developed proposing that computer programs be evaluated and verified by means of benchmark problems and solutions. These benchmark problems were to be developed independently by the staff, and submitted to applicants requesting that they provide solutions to these problems. The acceptability of an applicant's computer program would be determined by the similarity of the applicant's solutions and the benchmark solutions.

In October 1974, a work scope entitled, "Piping Benchmark Problems" was issued for assistance from a national laboratory in generating the benchmark solutions. This work scope described the requirements for such a program, and a preliminary list of problems suitable to be used as benchmarks. The Brookhaven National Laboratory in Upton, New York, was chosen to provide the required solution. In Fiscal Year 1975, the actual benchmark problems were selected by the NRC staff and BNL personnel, and computer programs that were to be used for generating the solutions were chosen and verified. Actual generation of benchmark solutions was begun in FY-1976. The computer program chosen for this effort was the program SAP-IV (Structural Analysis Program), developed at the University of California at Berkeley in the early 1970's and widely available.

Two reports detailing five benchmark problems and solutions were published in December 1977 (BNL-NUREG-21241-RS and BNL-NUREG-23645), and a draft request for information became available in January of 1978. The benchmark problems in these reports pertain to linear elastic structures and range from a simple structure under static loading to a two-loop primary piping system compiling a reactor vessel, steam generators, pumps and supports, subjected to earthquake motion. Additional benchmark problems have since been developed which pertain to elastic structures involving gaps (a non-linear problem). Other problems are being developed which include newer techniques, such as multiple support excitation, and preliminary efforts have been made in developing benchmarks for inelastic piping analysis.

In the course of licensing reviews, the NRC staff has required description and verification of structural programs since the early 1970's, and formalized these requirements in the Standard Review Plan published in 1975, (Section 3.9.1). Applicants submitted verification solutions which were based on simple benchmark problems only. The Piping Benchmark Program was designed to complement and expand these requirements and provide additional verification. However, methods of analysis of nuclear power plants for structural response under seismic and other loading conditions, which were the basis for these computer programs and were used in the design of early power plants (1968), have been presented in the open literature since the late 1960's.

applicants have also provided descriptions and verifications of their computer programs in the form of topical reports. One such topical report was submitted in 1976 by the Westinghouse Electric Co. titled: "Documentation of Selected Westinghouse Structural Analysis Computer Codes" (WCAP-8252). These programs and solutions were reviewed as thoroughly as possible without actually performing computer calculations, except for one program which involved a nonlinear analysis. The benchmark problem which the applicant submitted was reviewed under the Piping Benchmark Program by the BNL, by generating an independent solution to the same problem and confirming the applicant's results. (This problem will be incorporated in our standard list of benchmark problems.) Duke Power Co. also submitted verification of its method for structural analysis. The results by this applicant were also verified independently by BNL by running the same problems under the Piping Benchmark Program. A final report on this method will be published in the near future. Other analyses have been verified independently by the staff, and we are presently performing an evaluation and verification of the design techniques of certain component support members.

Related to the Benchmark Program is a much more general computer program evaluation project sponsored by the Armed Forces, and conducted by a group called the Interagency Software Evaluation Group (ISEG). The NRC staff is represented on this group. The objective of the group is to evaluate in depth the capabilities of some of the very large structural computer programs, such as ADINA, used nationwide.

STONE & WEBSTER ENGINEERING CORPORATION

ENGINEERING DIVISION MEMORANDUM NO. EMD-79-15 Rev. 1

ENGINEERING MECHANICS DIVISION

SUBJECT GENERAL PROCEDURE FOR THE STRESS ANALYSIS OF B31.1.0 BRANCH PIPING

DATE April 28, 1979

FROM RPWessel

TO Distribution

CC

1.0 PURPOSE

This procedure provides a uniform approach for the design/evaluation of branch piping that is consistent with USAS B31.1.0, 1967, through addenda to 1972 Code for pressure Piping. In delineating the various methods of analyzing branch piping, there is a latitude for independent judgement by the experienced stress analyst.

2.0 APPLICABILITY

Branch lines are explicitly addressed in the B31.1 Code stating that branch lines should be considered by applying correction factors (stress intensification factors) at the branch connection. It does not specify when a branch line can be analyzed with the run pipe or when it can be treated as a separate, uncoupled system.

In view of the above, branch piping connected to run piping shall not be included in the run pipe seismic model, in general, if the ratio of the moment of inertia of the run pipe to the branch pipe is greater than 10 to 1 (with certain exceptions as noted below).

3.0 MODELING PROCEDURE OF BRANCH PIPE

3.1 When the ratio of the moment of inertia of run pipe to branch pipe is equal to or <u>less than 10 to 1</u>:

3.1.1 The branch pipe should be modeled with the run pipe up to the first anchor on branch pipe (or up to and including the series of rigid constraints that effectively permits termination of the problem at some point remote from the pipe run). Piping outboard of the anchor (or series of constraints) should be analyzed by computer if the pipe is larger than 6" NPS and by manual methods if the pipe is 6" NPS and smaller.

3.2 When the ratio of the moment of inertia of run pipe to branch pipe is <u>more than 10 to 1</u>:

3.2.1 If the branch pipe is 6" NPS or smaller, the branch pipe should be decoupled from the run pipe and analyzed by the simplified manual method up to the first anchor (or up to and including the series of rigid constraints that effectively permits termination of the problem at some point remote from the pipe run).

If the branch pipe is larger than 6" NPS, the branch pipe may be decoupled from the run pipe and evaluated in the same manner as specified in this paragraph, except for using a computer analysis in lieu of the manual method.

3.2.2 Branch piping that is decoupled from the run pipe should be analyzed with the inclusion of the thermal and seismic movements of run pipe at the intersection of the run/branch point.

3.2.3 However, there are two <u>exceptions</u> to decoupling branch pipes as delineated in 3.2.1 and 3.2.2 above:

3.2.3.1 If an anchor or rigid constraint on the branch pipe is located near the run pipe and significantly restrains the movement of the run pipe, the branch run pipe should be included with the model of the run pipe, up to the anchor (or up to and including the series of rigid constraints that effectively permits termination of the problem at some point remote from the pipe run).

3.2.3.2 The branch pipe should be included in the
mathematical model of the run pipe if more
1 precise magnitude of reactions are required
at terminal points (i.e. equipment, penetrations
etc.) to determine their (the reactions)
acceptability.

4.0 <u>VALVES IN BRANCH LINES</u>

If the operational mode of valves located in branch piping causes a temperature change in the branch pipe, the temperature conditions must be considered in the branch pipe, regardless of the size of pipe or method of stress analysis used. This information should be obtained from the cognizant power engineer.

5.0 <u>LOADS ON SUPPORTS/CONSTRAINTS</u>

Where applicable, reactions from a computer analysis should be used for the design of supports. In the absence of computer generated reactions, the supports should be designed in accordance with the standard support loads of PS-4.

6.0 In all of the above cases, appropriate S.A.R. seismic qualification criteria should be applied, where applicable.

If there is a seismic class change in the branch pipe, the seismic analysis should include the piping outboard from the seismic class change to the first anchor (or up to and including the series of rigid constraints that effectively permits termination of the problem at some point remote from the pipe run. The non-seismic classified pipe from the first anchor (or from the series of constraints) may be analyzed by computer or manual calculations, depending on the diameter of pipe; typically, this is an instance where independent judgement must be exercised by the pipe stress analyst to determine where the non-seismic portion of the system should be started.

7.0 ATTACHMENT

Typical configuration of series of rigid constraints that effectively permits termination of problem. Attachment 1

R Klause for
R. P. Wessel

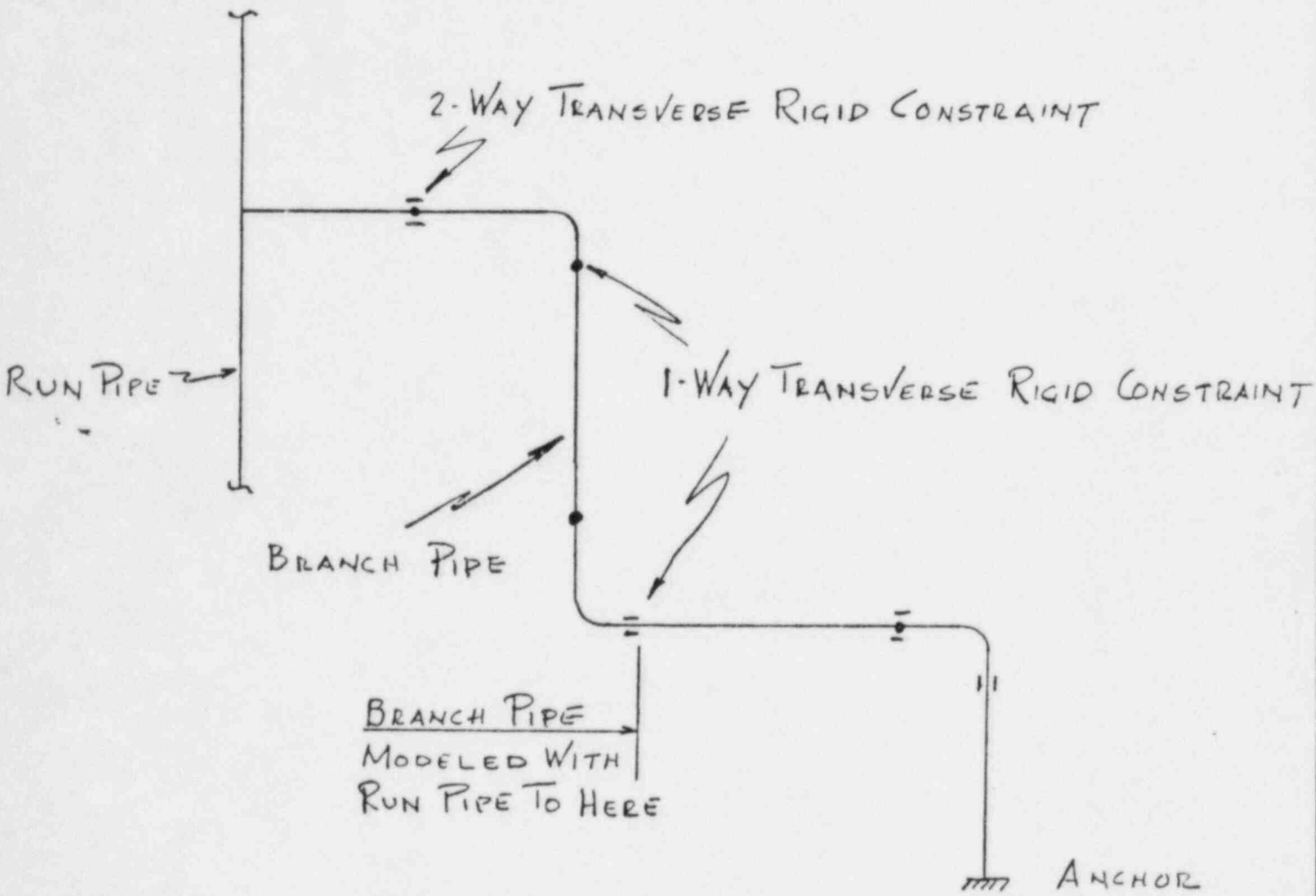
CALCULATION SHEET

STONE & WEBSTER ENGINEERING CORPORATION

J.O./W.O./CALCULATION NO.	REVISION	PAGE
EMD-79-25		

PREPARER/DATE	REVIEWER/CHECKER/DATE	INDEPENDENT REVIEWER/DATE

SUBJECT/TITLE	QA CATEGORY/CODE CLASS



A TYPICAL CONFIGURATION OF SERIES OF RIGID CONSTRAINTS THAT EFFECTIVELY PERMITS TERMINATION OF PROBLEM

ATTACHMENT 1

STONE & WEBSTER ENGINEERING CORPORATION

ENGINEERING DIVISION MEMORANDUM

NO. EMD-79-11
REV. 1

ENGINEERING MECHANICS DIVISION

SUBJECT STRESS INTENSIFICATION FACTORS AND STRESSES FOR REDUCED OUTLET BRANCH CONNECTIONS (B31.1)

DATE April 7, 1979

FROM RPWessel

CC

TO EMD TASK FORCE MEMBERS

1. Purpose

The purpose of this memorandum is to provide the acceptable code criteria for the calculation of Stress Intensification Factor (SIF) and stresses at the intersection of a run pipe and a reduced outlet branch connection. Branch connections covered here include: ANSI tees, unreinforced fabricated tees, reinforced fabricated tees, weldolets, sockolets, and branch connections per figure D-1 of Appendix D, Summer Addenda of 1973 to ANSI B31-1 Code, 1973. These factors are applicable to P-Stress, Nupipe, and hand calculations.

2. Procedures .

2.1 The following SIF should be applied on the branch pipe at the intersection of a run pipe and a reduced outlet branch connection:

a) SIF for branch connections listed below are provided in Attachments 1, 1A and 1B:
   - ANSI tees (type A)
   - unreinforced fabricated tees (type B)
   - reinforced fabricated tees, pad thickness same as run pipe thickness (type C)
   - reinforced fabricated tees, pad thickness equals 1.5 times run pipe thich (type D)

b) SIF for Weldolets - Attachment 2

c) SIF for sockolets - use the SIF for weldolets (per 2.1(b)) or use 1.3, whichever is greater.

   The SIF = 1.3 is the USAS B31.1.0 (1967) Code factor for fillet welds.

d) SIF for branch connections per fig. D-1, 1973 Summer Addenda ANSI B31.1 Code 1973 (see Attachment 3), where
   $r_m/R_m \leq 0.5$

$$i = 1.5\ (R_m/T_r)^{2/3}\ (r_m'/R_m)^{1/2}\ \ (T_b'/T_r)\ \ (r_m'/r_p)$$

2.2 For the calculation of stress at the intersection of a run pipe and a reduced outlet branch connection.

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It can be demonstrated that the following relationship will always give conservative values for the corrected branch stresses at reduced branch outlets:

$$S = S_o \left(\frac{i}{i_o} \times \frac{t_o}{t_s} \right)$$

where

S = Corrected Stress

S_o = Branch stress from SHOCK2 or 3

i = Stress intensification factor to be applied (from attachments 1, 1A, 1B, & 2)

i_o = Stress intensification factor from SHOCK 2 or 3 corresponding to S_o

t_o = Branch pipe thickness used in the SHOCK 2 or 3 calculation

t_s = The lesser of it_o or t_r

t_r = Nominal thickness of the run pipe

In cases where the value of S exceeds the allowable stress, the exact expression from which the above relation is derived should be applied. This exact expression is:

$$S = S_o \left[\frac{(i)^2 + (a)^2}{(i_o)^2 + (a)^2} \right]^{1/2} \frac{Z_o}{\pi \quad r^2 t_s}$$

where

a = The ratio of torsional moment to bending moment (M_T/M_b) in the branch pipe. These values are extracted from the SHOCK 2 or 3 run.

r = The mean radius of the branch pipe.

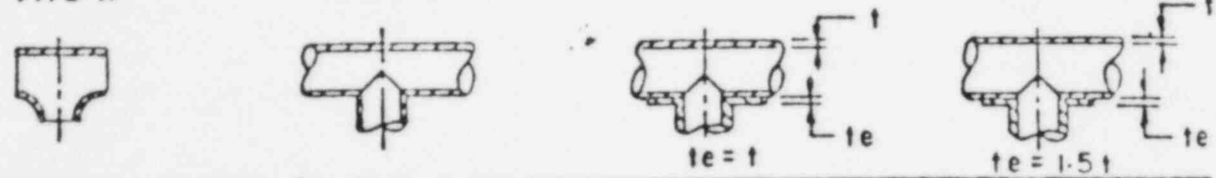
Z_o = The section modulus of the branch pipe extracted from SHOCK 2 or 3

other notations are the same as before

J. H. Bolling for

RPWessel

TYPE A TYPE B TYPE C TYPE D



3" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·120	1·9542	5·2471	2·6695	2·0646
20					
30					
STD	·216	1·2956	3·4787	1·7698	1·3688
40	·216	1·2956	3·4787	1·7698	1·3688
60					
X STR	·300	1·0230	2·7468	1·3974	1·0808
80	·300	1·0230	2·7468	1·3974	1·0808
100					
120					
140					
160	·437	1·0	2·0763	1·0564	1·0
XX STR	·600	1·0	1·6203	1·0	1·0

4" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·120	2·3229	6·2373	3·1733	2·4543
20					
30					
STD	·237	1·4496	3·8922	1·9802	1·5315
40	·237	1·4496	3·8922	1·9802	1·5315
60					
X STR	·337	1·1284	3·0298	1·5415	1·1922
80	·337	1·1284	3·0298	1·5415	1·1922
100					
120	·437	1·0	2·5070	1·2755	1·0
140					
160	·531	1·0	2·1669	1·1024	1·0
XX STR	·674	1·0	1·8040	1·0	1·0

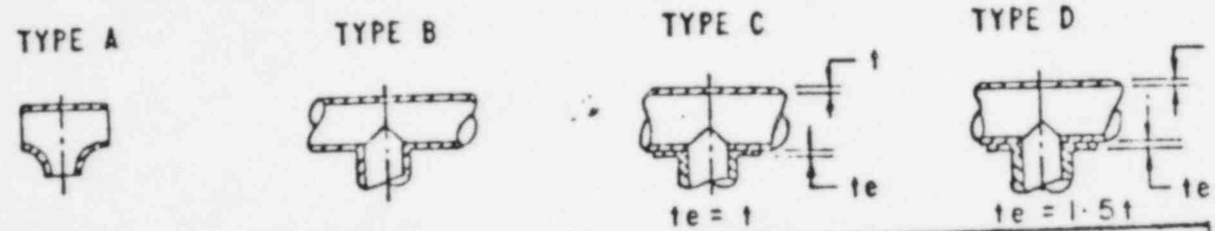
6" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·134	2·8049	7·5315	3·8317	2·9635
20					
30					
STD	·280	1·6904	4·5389	2·3092	1·7860
40	·280	1·6094	4·5389	2·3092	1·7860
60					
X STR	·432	1·2457	3·3449	1·7018	1·3162
80	·432	1·2457	3·3449	1·7018	1·3162
100					
120	·562	1·0307	2·7675	1·4080	1·0889
140					
160	·718	1·0	2·3100	1·1752	1·0
XX STR	·864	1·0	2·0079	1·0215	1·0

8" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·148	3·1365	8·4220	4·2847	3·3139
20	·250	2·1936	5·8901	2·9966	2·3176
30	·277	2·0443	5·4892	2·7927	2·1593
STD	·322	1·8425	4·9473	2·5170	1·9467
40	·322	1·8425	4·9473	2·5170	1·9467
60	·406	1·5679	4·2101	2·1419	1·6566
X STR	·500	1·3544	3·6366	1·8502	1·4309
80	·500	1·3544	3·6366	1·8502	1·4309
100	·593	1·1996	3·2211	1·6389	1·2675
120	·718	1·0450	2·8058	1·4275	1·1040
140	·812	1·0	2·5644	1·3047	1·0017
160	·906	1·0	2·3645	1·2030	1·0
XX STR	·875	1·0	2·4268	1·2347	1·0

BASED ON RUN PIPE

ATTACHMENT 1

POWER INDUSTRY GROUP		STRESS ITENSIFICATION FACTORS FOR TEE JOINTS SHEET 1	1	ORIGINAL
CHECKED	CSJ 11-20-72		ISSUE	DESCRIPTION
CORRECT	CSK 12-20-72			
APPROVED	[illegible]	STANDARD DESIGN GUIDE	STD-MSA-1500-1-	



10" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·165	3·3823	9·0819	4·6205	3·5735
20	·250	2·5508	6·8491	3·4946	2·6950
30	·307	2·2164	5·9512	3·0277	2·3417
STD	·365	1·9676	5·2831	2·6878	2·0788
40	·365	1·9676	5·2831	2·6878	2·0788
60	·500	1·5811	4·2454	2·1599	1·6705
X STR	·500	1·5811	4·2454	2·1599	1·6705
80	·593	1·4027	3·7664	1·9162	1·4820
100	·718	1·2248	3·2886	1·6731	1·2940
120	·843	1·0914	2·9304	1·4909	1·1531
140	1·000	1·0	2·5872	1·3163	1·0180
160	1·125	1·0	2·3716	1·2066	1·0
XX STR					

12" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·180	3·5797	9·6120	4·8902	3·7821
20	·250	2·8651	7·6932	3·9140	3·0271
30	·330	2·3708	6·3658	3·2387	2·5048
STD	·375	2·1720	5·8321	2·9672	2·2948
40	·406	2·0565	5·5218	2·8093	2·1727
60	·562	1·6417	4·4082	2·2427	1·7345
X STR	·500	1·7803	4·7817	2·4327	1·8815
80	·687	1·4263	3·8296	1·9484	1·5069
100	·843	1·2337	3·3125	1·6853	1·3034
120	1·000	1·0912	2·9299	1·4906	1·1529
140	1·125	1·0021	2·6907	1·3689	1·0587
160	1·312	1·0	2·4012	1·2217	1·0
XX STR					

14" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·250	3·0530	8·1975	4·1705	3·2256
20	·312	2·6260	7·0512	3·5374	2·7745
30	·375	2·3159	6·2183	3·1636	2·4468
STD	·375	2·3159	6·2183	3·1636	2·4468
40	·437	2·0850	5·5984	2·8482	2·2029
60	·593	1·6879	4·5321	2·3057	1·7833
X STR	·500	1·9000	5·1016	2·5955	2·0074
80	·750	1·4323	3·8459	1·9567	1·5133
100	·937	1·2229	3·2836	1·6706	1·2921
120	1·052	1·1178	3·0014	1·5270	1·1810
140	1·250	1·0	2·6662	1·3565	1·0491
160	1·406	1·0	2·4450	1·2439	1·0
XX STR					

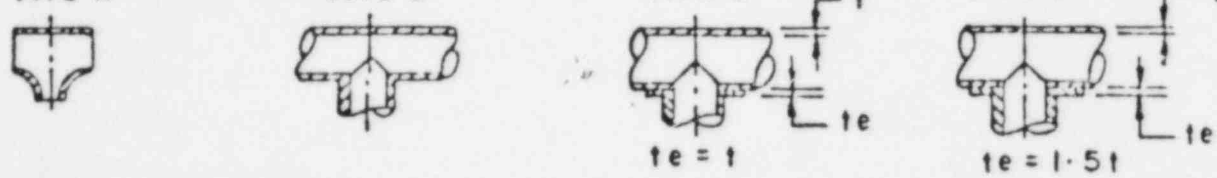
16" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·250	3·3423	8·9745	4·5658	3·5313
20	·312	2·8758	7·7219	3·9286	3·0384
30	·375	2·5373	6·8130	3·4662	2·6808
STD	·375	2·5373	6·8130	3·4662	2·6808
40	·500	2·0833	5·5939	2·8460	2·2011
60	·656	1·7266	4·6360	2·3586	1·8242
X STR	·500	2·0833	5·5939	2·8460	2·2011
80	·843	1·4490	3·8907	1·9794	1·5309
100	1·031	1·2564	3·3736	1·7164	1·3275
120	1·218	1·1149	2·9937	1·5231	1·1780
140	1·437	1·0	2·6549	1·3507	1·0446
160	1·562	1·0	2·4967	1·2703	1·0
XX STR					

BASED ON RUN PIPE

ATTACHMENT 1

POWER INDUSTRY GROUP		STRESS INTENSIFICATION FACTORS FOR TEE JOINTS SHEET 2	1	[illegible]
CHECKED	[illegible] 12-20-72		ISSUE	DESCRIPTION
CORRECT	[illegible] 12-20-72			
APPROVED	[illegible]	STANDARD DESIGN GUIDE	STD-MSA-1509-2	
ISSUE	2	3	4	5

TYPE A TYPE B TYPE C TYPE D



18" NOMINAL RUN SIZE						20" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D	SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·250	3·6196	9·7191	4·9446	3·8242	10	·250	3·8864	10·4355	5·3091	4·1061
20	·312	3·1153	8·3648	4·2557	3·2914	20	·375	2·9536	7·9307	4·0348	3·1206
30	·437	2·4767	6·6503	3·3834	2·6168	30	·500	2·4277	6·5187	3·3165	2·5650
STD	·375	2·7492	7·3819	3·7556	2·9046	STD	·375	2·9536	7·9307	4·0348	3·1206
40	·562	2·0846	5·5972	2·8476	2·2024	40	·593	2·1560	5·7997	2·9507	2·2821
60	·750	1·7074	4·5846	2·3324	1·8039	60	·812	1·7384	4·6679	2·3748	1·8367
X STR	·500	2·2589	6·0653	3·0858	2·3656	X STR	·500	2·4277	6·5187	3·3165	2·5650
80	·937	1·4613	3·9238	1·9963	1·5439	80	1·031	1·4714	3·9509	2·0100	1·5546
100	1·157	1·2587	3·3797	1·7195	1·3299	100	1·250	1·2841	3·4479	1·7541	1·3567
120	1·343	1·1312	3·0375	1·5453	1·1952	120	1·500	1·1270	3·0260	1·5395	1·1907
140	1·562	1·0139	2·7224	1·3850	1·0712	140	1·750	1·0078	2·7059	1·3767	1·0647
160	1·750	1·0	2·5045	1·2742	1·0	160	1·937	1·0	2·5116	1·2778	1·0
XX STR						XX STR					

24" NOMINAL RUN SIZE						30" NOMINAL RUN SIZE					
SCH	t	TYPE A	TYPE B	TYPE C	TYPE D	SCH	t	TYPE A	TYPE B	TYPE C	TYPE D
10	·250	4·3949	11·8008	6·0037	4·6434	10	·312				
20	·375	3·3422	8·9741	4·5657	3·5311						
30	·562	2·5388	6·8168	3·4681	2·6823						
STD	·375	3·3422	8·9741	4·5657	3·5311	STD	·375				
40	·687	2·2128	5·9417	3·0229	2·3379		·438				
60	·937	1·7876	4·7999	2·4420	1·8887	60					
X STR	·500	2·7493	7·3822	3·7558	2·9048	X STR	·500	3·21	8·57	4·371	3·382
80	1·218	1·4876	3·9942	2·0321	1·5717		·562				
100	1·500	1·2841	3·4479	1·7541	1·3567		·625				
120	1·750	1·1501	3·0881	1·5711	1·2151						
140	2·062	1·0824	2·9053	1·4786	1·1436						
160	2·312	1·0	2·5215	1·2828	1·0						
XX STR											

BASED ON RUN PIPE

ATTACHMENT 18

POWER INDUSTRY GROUP		STRESS INTENSIFICATION FACTORS FOR TEE JOINTS SHEET 3	1	
CHECKED	[illegible] 12-23-76			
CORRECT	[illegible] 12-23-76		ISSUE	DESCRIPTION
APPROVED	[illegible]	STANDARD DESIGN GUIDE	STD-MSA-1509-3-	
ISSUE	2	3 4 5		

CALCULATION SHEET

J.O./W.O./CALCULATION NO.	REVISION	PAGE
516.05.04-NP(B)-013		2

PREPARER/DATE	REVIEWER/CHECKER/DATE	INDEPENDENT REVIEWER/DATE
4-6-79	JMC 4/6/79	4/6 79

SUBJECT/TITLE: Stress Intensification Factors For "Weldolets"

QA CATEGORY/CODE CLASS

Stress Intensification Factors
Bonney Forge Corporation
Weldolets

Standard Wt			Extra Strong		
Run Size NPS	Branch Size	SIF	Run Size NPS	Branch Size	SIF
2	All	1.52	2	All	1.18*
3	↓	1.57	3	↓	1.24
4		1.76	4		1.37
6		2.05	6		1.51
8		2.23	8		1.64
10		2.38	10		1.92
12		2.63	12		2.16
14		2.81	14		2.30
16		3.07	16		2.52
18		3.33	18		2.74
20		3.58	20		2.94
22		3.82	22		3.14
24		4.05	24		3.33
30		4.71	30		3.88
36	↓	5.33	36	↓	4.39

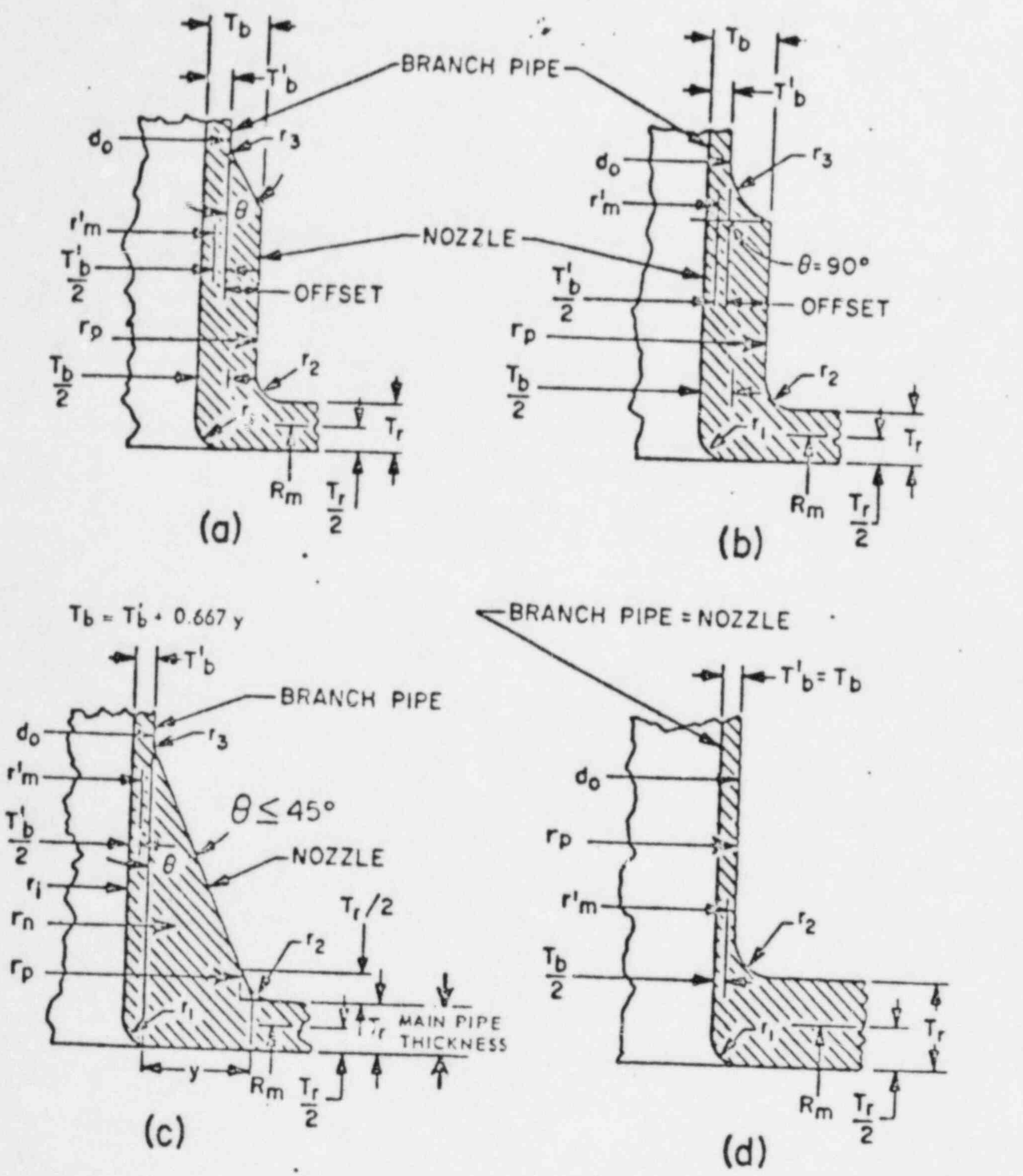


FIG. D-1 NOZZLE DIMENSIONS

ATTACHMENT 3